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LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

1980 ANNUAL REPORT

(October 1, 1979 - September 30, 1980)

Hanford Engineering Development Laboratory

Prepared by
W.N. McElroy, F.B.K. Kam,
E.D. McGarry, and A. Fabry

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Prepared for the U.S. Nuclear Regulatory Commission

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Hanford Engineering Development Laboratory

Operated by Westinghouse Hanford Company
P.O. Box 1970 Richland, WA 99352
A Subsidiary of Westinghouse Electric Corporation

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LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

1980 Annual Report

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LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM

1980 ANNUAL REPORT

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ABSTRACT

This report describes progress made in the Light Water Reactor Pressure Vessel Surveillance Dosimetry Improvement Program during 1980. The primary concern of this program is to improve, standardize, and maintain dosimetry, damage correlation, and the associated reactor analysis procedures used for predicting the integrated effect of netutron exposure to LWR pressure vessels and support structures.

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FOREWORD

The light water reactor pressure vessel (LWR-PV) surveillance dosimetry improvement program has been established by NRC in recognition of the importance of improving, maintaining, and standardizing neutron dosimetry, damage correlation, and the associated reactor analysis data and procedures used for predicting the integrated effect of neutron exposure to LWR pressure vessels. A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links among the USNRC supported activities at HEDL, ORNL, NBS and NRL and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Jülich, Germany) and several UK laboratories have been extended to a number of other countries and laboratories. These cooperative links have been strengthened by the active membership of the scientific staff of many of the participating countries and laboratories in the ASTM E10 Committee on Nuclear Technology and Applications. Several subcommittees of ASTM E10 are responsible for the preparation of LWR-PV surveillance standards.

The primary objective of the multilaboratory program is to prepare an updated and improved set of dosimetry, damage correlation, and associated reactor analysis ASTM Standards for LWR-PV irradiation surveillance programs. Supporting this objective are a series of analytical and experimental validation and calibration studies in "Standard, Reference, and Controlled Environment Benchmark Fields," reactor "Test Regions," and operating power reactor "Surveillance Positions."

These studies will establish and certify the precision and accuracy of the measurement and predictive methods which are recommended for use in the ASTM Standards. Consistent and accurate measurement and data analysis techniques and methods, therefore, will have been developed and validated along with guidelines for required neutron field calculations that are used to correlate changes in material properties with the characteristics of the neutron radiation field. It is expected that the application of the established

ASTM Standards will permit the reporting of measured materials property changes and neutron exposures to an accuracy and precision within bounds of 10 to 30%, depending on the measured metallurgical variable and neutron environment.

The assessment of the radiation-induced degradation of material properties in a power reactor pressure vessel requires accurate definition of the neutron field from the outer region of the reactor core to the outer boundaries of the pressure vessel. Problems with measuring neutron flux and spectrum are associated with two distinct components of LWR-PV irradiation surveillance procedures: (1) proper application of calculational estimates of the neutron fluence delivered to in-vessel surveillance positions, various locations in the vessel wall, and ex-vessel support structures and surveillance positions, and (2) understanding the relationship between material property changes in reactor vessels, in-vessel support structures, and in metallurgical test specimens in test reactors and at accelerated neutron flux positions in operating power reactors.

The first component requires validation and calibration experiments in a variety of neutron irradiation test facilities including LWR-PV mock-ups, power reactor surveillance positions, and related benchmark neutron fields. The benchmarks serve as a permanent reference measurement for neutron flux and fluence detection techniques, which are continually under development and widely applied by laboratories with different levels of capability. The second component requires a serious extrapolation of an observed neutron induced mechanical property change from test reactor "test regions" and operating power reactor "surveillance positions" to locations inside the body of the pressure vessel wall and to ex-vessel support structures. The neutron flux at the vessel inner wall is up to one order of magnitude lower than at surveillance specimen positions and up to two orders of magnitude lower than for test reactor positions. At the vessel outer wall, the neutron flux is one order of magnitude or more lower than at the vessel inner wall. Further, the neutron spectrum at, within, and leaving the vessel is substantially different.

In order to meet the reactor pressure vessel radiation monitoring requirements, a variety of neutron flux and fluence detectors are employed, most of which are passive. Each detector must be validated for application to the higher flux and harder neutron spectrum of the test reactor "test region" and to the lower flux and degraded neutron spectrum at "surveillance positions." Required detectors must respond to neutrons of various energies so that multigroup spectra can be determined with accuracy sufficient for adequate damage response estimates. Proposed detectors for the program include radiometric detectors, helium accumulation fluence monitors, solid state track recorders, and damage monitors.

The necessity for pressure vessel mock-up facilities for dosimetry investigations and for irradiation of metallurgical specimens was recognized early in the formation of the NRC program. Experimental studies associated with high and low flux versions of a PWR pressure vessel mock-up are in progress. The low flux version is known as the Poolside Critical Assembly (PCA) and the high flux version is known as the Pool Side Facility (PSF). Both are located at ORNL. As specialized benchmarks, these facilities will provide well-characterized neutron environments where active and passive neutron dosimetry, various types of LWR-PV neutron field calculations, and temperature-controlled metallurgical specimen exposures are brought together.

The results of the measurement and calculational strategies outlined here will be made available for use by the nuclear industry as ASTM Standards. Federal Regulation 10CFR50 already calls for adherence to several ASTM Standards which require establishment of a surveillance program for each power reactor and incorporation of flux monitors and neutron field evaluation. Revised and new standards in preparation will be carefully structured to be up-to-date, flexible, and, above all, consistent.

In order to meet the reactor pressure vessel radiation monitoring requirements, a variety of neutron flux and fluence detectors are employed, most of which are passive. Each detector must be validated for application to the higher flux and harder neutron spectrum of the test reactor "test region" and to the lower flux and softened neutron spectrum of "surveillance positions." Reaction rates must account for neutrons of various energies [illegible] with spectral coefficient for [illegible] through neutron [illegible]. The dosimetry for the program include radiometric monitors, helium accumulation fluence monitors, solid state track recorders, and damage monitors.

The need for precise dosimetry facilities for the necessary investigations and the development of techniques that have been recognized early in the formation of the NRC program. [illegible] at ORNL and the [illegible] at NBS. [illegible] these facilities will provide well-characterized [illegible] environments for the [illegible] benchmarks [illegible].

The results of the measurement and interpretation [illegible] will be made available for use by the nuclear industry in both [illegible] Federal Regulation 10CFR50 already calls for adherence to several ASTM standards which require [illegible] a surveillance program for each power reactor and interpretation of flux, fluence and neutron field evaluations. Revisions and any changes in presentation will be carefully structured so as to be easy, flexible, and, above all, consistent.

LWR PRESSURE VESSEL SURVEILLANCE DOSIMETRY IMPROVEMENT PROGRAM
1980 ANNUAL REPORT

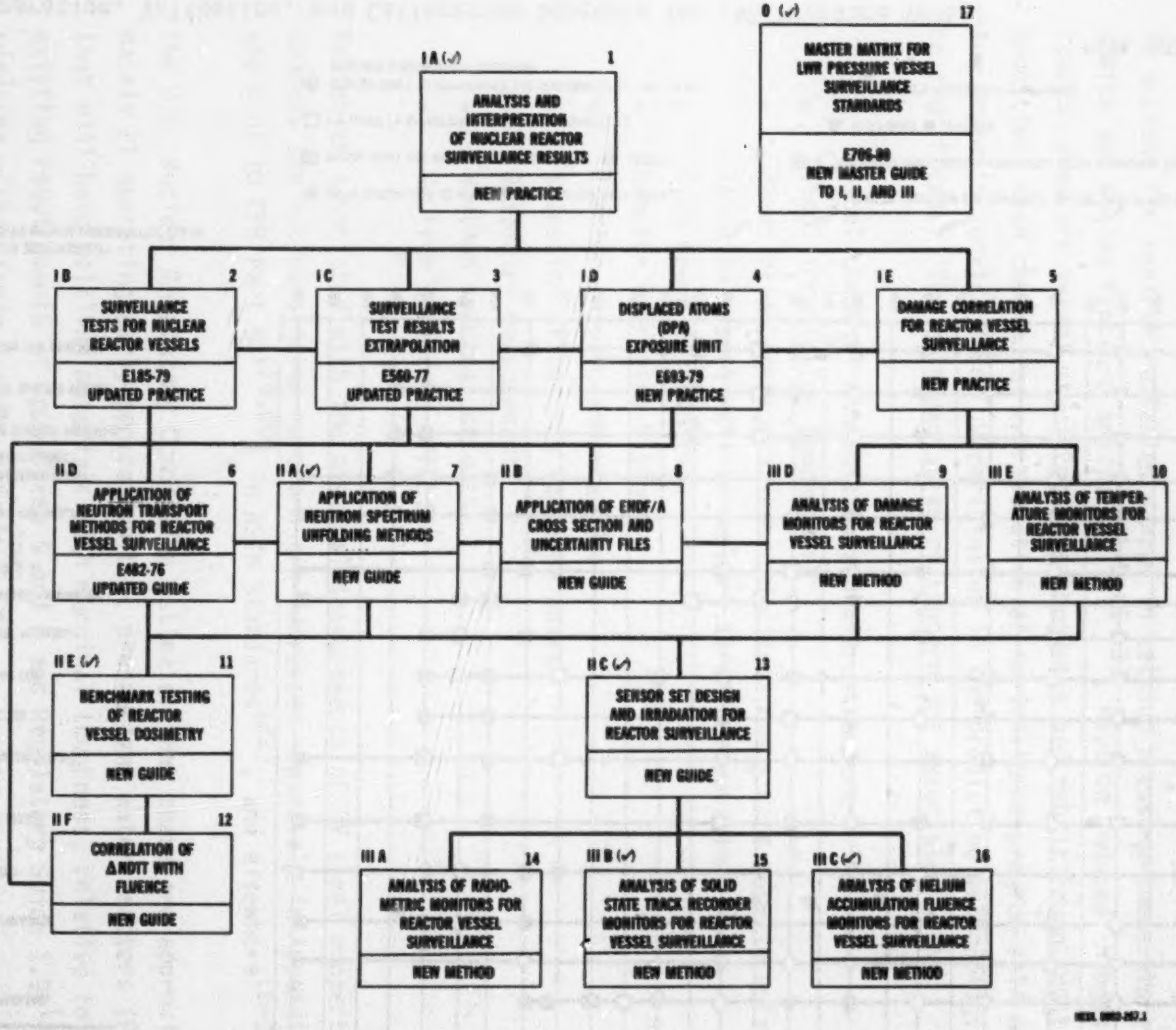
I. INTRODUCTION

Aging light water reactor pressure vessels (LWR-PV) are accumulating significant neutron fluence exposures, with consequent changes in their steel fracture toughness and embrittlement characteristics. Recognizing that accurate and validated measurement and data analysis procedures are needed to periodically evaluate the metallurgical condition of these reactor vessels, the U. S. Nuclear Regulatory Commission (NRC) has established the LWR-PV Surveillance Dosimetry Improvement Program. The primary concern of this program is to improve, standardize, and maintain dosimetry, damage correlation, and the associated reactor analysis procedures used for predicting the integrated effect of neutron exposure to LWR pressure vessels and support structures.

A vigorous research effort attacking the same measurement and analysis problems exists worldwide, and strong cooperative links between the USNRC supported activities at HEDL, ORNL, NBS and NRL and those supported by CEN/SCK (Mol, Belgium), EPRI (Palo Alto, USA), KFA (Jülich, Germany) and several UK laboratories have been extended to a number of other countries and laboratories.(1-7) These cooperative links have been strengthened by the active membership of the scientific staff of many of the participating countries and laboratories in the ASTM E10 Committee on Nuclear Technology and Applications. Several subcommittees of ASTM E10 are responsible for the preparation of LWR-PV surveillance standards.

The major benefit of this program will be a significant improvement in the accuracy of the assessment of the remaining safe operating lifetime of light water reactor pressure vessels and support structures. A primary objective of the multilaboratory program is to prepare an updated and improved set of dosimetry, damage correlation, and the associated reactor analysis ASTM standards for LWR-PV irradiation surveillance programs, as described in

Figures 1 and 2. Supporting this objective are a series of analytical and experimental validation and calibration studies in "Benchmark Neutron Fields," reactor "Test Regions," and operating power reactor "Surveillance Positions." These studies will establish and certify the precision and accuracy of the measurement and predictive methods which are recommended for use in these standards. Consistent and accurate measurement and data analysis techniques and methods, therefore, will have been developed and validated along with guidelines for required neutron field calculations. These calculations, in turn, are used 1) to correlate changes in material properties with the characteristics of the neutron radiation field and 2) to predict end-of-life (EOL) pressure vessel and support structure steel toughness and embrittlement from power and test reactor data.


I A (✓) 1
ANALYSIS AND INTERPRETATION OF NUCLEAR REACTOR SURVEILLANCE RESULTS
NEW PRACTICE
0 (✓) 17
MASTER MATRIX FOR LWR PRESSURE VESSEL SURVEILLANCE STANDARDS
E706-80
NEW MASTER GUIDE TO I, II, AND III
I B 2
SURVEILLANCE TESTS FOR NUCLEAR REACTOR VESSELS
E185-79
UPDATED PRACTICE
I C 3
SURVEILLANCE TEST RESULTS EXTRAPOLATION
E560-77
UPDATED PRACTICE
I D 4
DISPLACED ATOMS (DPA) EXPOSURE UNIT
E693-79
NEW PRACTICE
I E 5
DAMAGE CORRELATION FOR REACTOR VESSEL SURVEILLANCE
NEW PRACTICE
II D 6
APPLICATION OF NEUTRON TRANSPORT METHODS FOR REACTOR VESSEL SURVEILLANCE
E482-76
UPDATED GUIDE
II A (✓) 7
APPLICATION OF NEUTRON SPECTRUM UNFOLDING METHODS
NEW GUIDE
II B 8
APPLICATION OF ENDF/A CROSS SECTION AND UNCERTAINTY FILES
NEW GUIDE
III D 9
ANALYSIS OF DAMAGE MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD
III E 10
ANALYSIS OF TEMPERATURE MONITORS FOR REACTOR VESSEL SURVEILLANCE
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II E (✓) 11
BENCHMARK TESTING OF REACTOR VESSEL DOSIMETRY
NEW GUIDE
II F 12
CORRELATION OF ΔNDTT WITH FLUENCE
NEW GUIDE
II C (✓) 13
SENSOR SET DESIGN AND IRRADIATION FOR REACTOR SURVEILLANCE
NEW GUIDE
III A 14
ANALYSIS OF RADIOMETRIC MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD
III B (✓) 15
ANALYSIS OF SOLID STATE TRACK RECORDER MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD
III C (✓) 16
ANALYSIS OF HELIUM ACCUMULATION FLUENCE MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD

FIGURE 1. Interrelationships for ASTM Standards for Surveillance of Nuclear Reactor Pressure Vessels.
Neg 801099-1

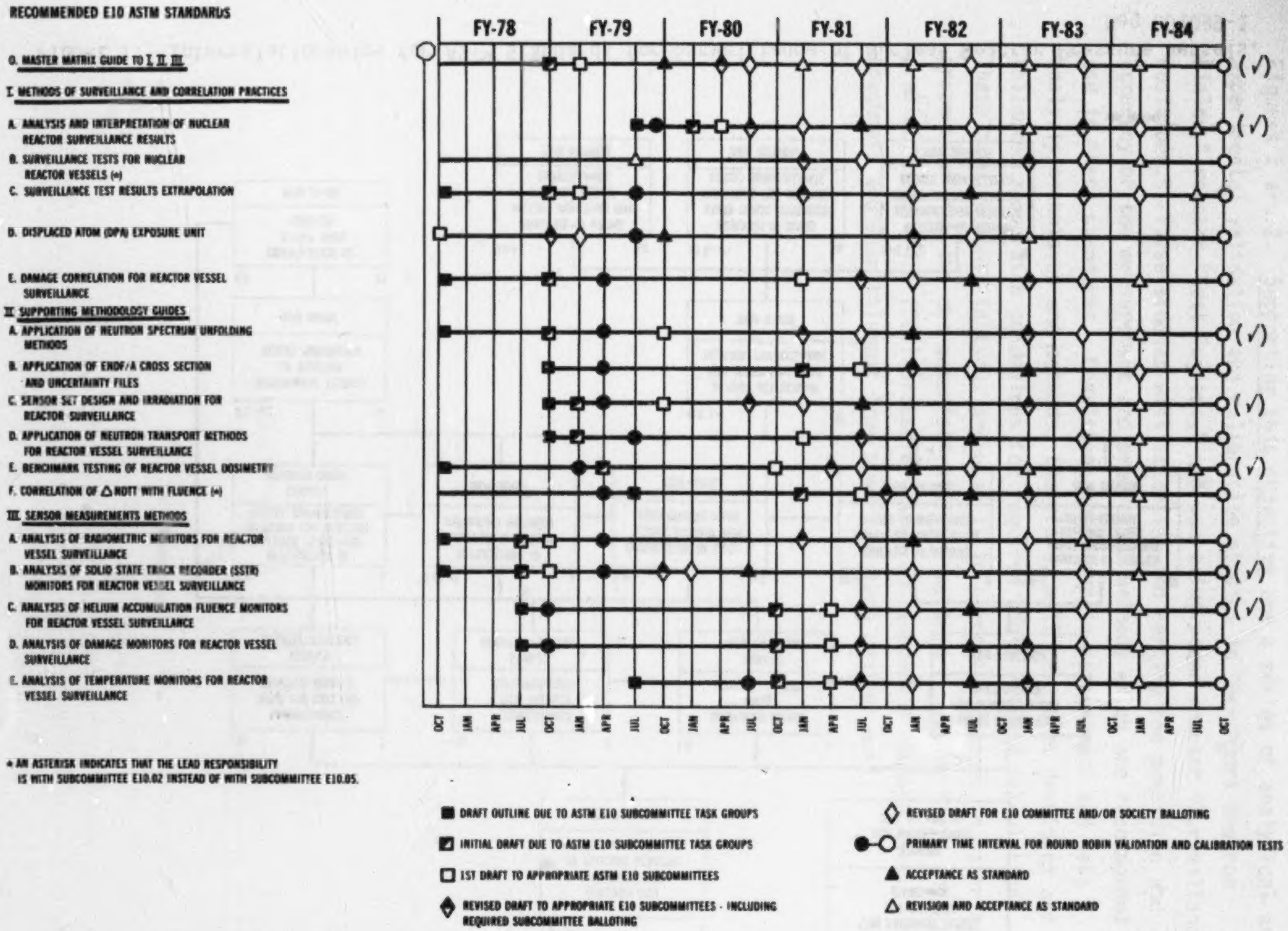

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FIGURE 2. Preparation, Validation, and Calibration Schedule for LWR Pressure Vessel Surveillance Standards. Neg 8010270-11

II. FY 1980 RESEARCH RESULTS - SUMMARY

To account for neutron radiation damage in setting pressure-temperature limits and making fracture analysis(1-3, 6-23), neutron induced changes in reactor pressure vessel steel fracture toughness and embrittlement must be predicted, then checked by extrapolation of surveillance program data during the vessel's service life. Uncertainties in the predicting methodology can be significant. The main variables of concern are associated with:

- Steel chemical composition and microstructure
- Steel irradiation temperature
- Power plant configurations and dimensions - core edge to surveillance to vessel wall to support structure positions
- Core power distribution
- Reactor operating history
- Reactor physics computations
- Selection of neutron exposure units
- Dosimetry measurements
- Neutron spectral effects
- Neutron dose rate effects.

Variables associated with the physical measurements of PV steel property changes are not considered here and are addressed separately in Appendices G and H of 10 CFR Part 50(19,20) in ASTM Standards(22), and elsewhere(6-17).

The U.S. Nuclear Regulatory Commission estimates that there are approximately 21 operating early generation U.S. pressurized water reactors (PWR) that will have beltline materials with marginal toughness, relative to the existing requirements of Appendices G and H and Regulatory Guide 1.99(21) sometime within their service life.(24)

As older vessels become more highly irradiated, the predictive capability for changes in fracture toughness and embrittlement must improve. Since during the vessel's service life an increasing amount of information will be available from test reactor and surveillance programs, better procedures to evaluate and use this information can and must be developed. The most appropriate way to make these procedures available is through voluntary consensus standards, such as those now being developed by ASTM Committee E10 on Nuclear Technology and Applications and discussed here.

Important summary highlights of FY 80 research activities of this multilaboratory program are:

A. The preparation of first, revised, or final drafts, Figures 1 and 2, of seven (✓) of seventeen ASTM standards which focus on the critical neutron exposure (dosimetry), damage analysis (data correlation), and the associated reactor analysis and interpretation aspects of the problem of guaranteeing the safety and integrity of the pressure vessel boundary of LWR power reactors.(23)

B. Initiation and completion of important supporting validation and calibration studies, reviews, and neutron field experimentation, Table 1, which demonstrate and certify the direct applicability of the sixteen ASTM standards (five "practices," six "guides," and five "methods").(2,4,5,25-27)

C. The completion of key experimental dosimetry/physics studies associated with the PCA low flux version of a PWR pressure vessel mockup, Figures 3-6 and Tables 2-6.(2,4,25-27)

D. The completion of a "Blind Testing" program using the PCA results to assist in the verification of the reliability of neutron physics calculations that are used for predicting PV steel toughness and embrittlement from power reactor surveillance data. The status of this effort is reviewed in a separate paper presented at the NRC 8th Safety Research Information meeting.(4) [Compared with measurements, most "Blind Test" calculations showed a trend towards underestimating the fluxes 5-25% and more so (50% and greater) for the cylindrical geometry. Only the one-dimensional calculations tended to overestimate the fluxes. If the same trend were found in the calculation of commercial power reactors, then the extrapolation from surveillance measurements to the interior of the pressure vessel could lead to significant nonconservative errors in flux, fluence, and damage estimates.]

E. The completion of the testing and the start in early 1980 of the Oak Ridge Research Reactor irradiaton of the metallurgy/dosimetry test assembly for the PSF experiment, which is a high flux version of a PWR pressure vessel mockup, Figures 9-23 and Tables 7 to 11.

F. The completion of initial work associated with the evaluation and reevaluation of exposure units and values for existing and new metallurgical data bases (Reg. Guide 1.99, MPC, EPRI, and others), Figures 7,8, and 24 and Tables 12-17. [This initial work involved the reanalysis of data from 19 PWR surveillance capsule reports for Westinghouse power plants. Using a consistent set of auxiliary data and dosimetry adjusted reactor physics results, the revised fluence values for E > 1 MeV averaged 42% higher than the originally reported values. The range of fluence values (new/old) was from a low of 1.08 to a high of 2.11. This indicates a definite need to re-evaluate other power reactor surveillance and test reactor exposure data.]

TABLE 1

LIST OF NEUTRON FIELDS FOR LWR-PV DOSIMETRY VALIDATION AND CALIBRATION STUDIES

Neutron Field	Type of Dosimetry*	Status
Benchmarks for Calibration		
• ^{252}Cf and ^{235}U Fission Neutron Irradiation Facilities at NBS (Standard)	Sensor calibrations	Preparation of fluence counting standards in progress for participating laboratories. ^{252}Cf is absolute fast neutron fluence standard.
• ^{235}U Fission Field at CEN/SCK (Standard)	Sensor calibrations	Applied routinely to RM, DM and other sensor calibrations for PCA and other neutron field measurements.
• ISNF Irradiation Facilities at NBS (Standard)	Sensor calibrations (particularly detectors with response range below 0.5 MeV)	Fission rate ratios established with fission chambers. Application to RM and other sensors planned.
• CRMF Irradiation Facility at EG&G (Reference)	Sensor calibrations	Planning stage.
• PCA/PSF Irradiation Facilities at ORNL (Reference)	Sensor validations and surveillance perturbation investigations	BSR-SDMF being established at ORNL.
Benchmarks for Transport Calculation Validation		
• Iron Shells Field at CEN/SCK (Reference)	Active and passive RM, SSTR, NE	CEN/SCK spectrometry in progress. NBS fission chamber measurements completed. Planning stage for passive sensors.
• PCA at ORNL (Reference)	Active and passive RM, SSTR, NE	Radiometric, fission chamber, track recorder, and neutron and gamma spectrometry measurements in progress. Results used for "Blind Test" program.
Test Regions for Dosimetry Method** Validation		
• BSR-HSST Dosimetry-Test (Test Reactor at ORNL)	Radiometric monitors (RM)	Irradiation and dosimetry counting completed. Analysis of Interlaboratory dosimetry in progress.
• FRJ-1 and FRJ-2 (Test Reactor at KFA, Germany)	RM, HAFM, TM (Melt Wires), DM (Quartz)	"Test Region" irradiations in metallurgical rigs in progress. Interlaboratory dosimetry analysis in progress.
• DIDO, PLUTO, and HERALD (Test Reactors in UK)	RM, HAFM, TM, DM	Planning stage, UK results are available.
• Buffalo NRL-NRC (Test Reactor at University of Buffalo)	Radiometric monitors	Planning stage, NRL results are available.
• University of Virginia (Test Reactor)	Radiometric monitors	Planning stage, Westinghouse results are available.
• BR-3 (PWR Power Reactor at CEN/SCK, Mol, Belgium)	RM, HAFM, DM (Quartz), TM (Melt Wires)	In- and Ex-Vessel "Test Region" irradiations in progress.
• Arkansas Power and Light Company PWR Unit #1 (Russelville, Arkansas)	Radiometric monitors	Ex-vessel "Test Region" irradiations and sensor counting and analysis in progress.
• Garigliano Reactor (BWR Power Reactor at Rome, Italy)	Radiometric monitors	In-vessel "Test Region" irradiation completed and analysis in progress.
• Brown's Ferry 3 (BWR)	RM, SSTR, Ionization Chambers	In- and ex-vessel "Test Region" irradiations completed and analysis in progress.
• McGuire I (PWR)	RM, SSTR, Proton Recoil	Ex-vessel "Test Region" irradiations planned and scheduled.

*RM: Radiometric Monitors
SSTR: Solid State Track Recorders
HAFM: Helium Accumulation Fluence Monitors
NE: Nuclear Emulsions
DM: Damage Monitors
TM: Temperature Monitors
BSR-SDMF: ORNL Bulk Shielding Reactor - Surveillance Dosimetry Measurement Facility.

**These neutron fields are also being used for physics and metallurgical damage correlation procedures and data validation and calibration studies.

III. ASTM STANDARDS FOR SURVEILLANCE OF NUCLEAR REACTOR PRESSURE VESSELS

The status of the preparation of the seventeen ASTM standards was very briefly summarized in Section II. The current schedule for the preparation, acceptance, and revision of all of the new and updated standards is shown in Figure 2. A summary listing of the type and the status of neutron field studies associated with the validation and calibration of the 16 ASTM Standards is provided in Table 1. The ASTM Master Matrix Standard, Appendix A, provides more detailed information on the current 1) scope, 2) areas of application, and 3) interrelationships for the set of sixteen standards.(23)

With reference to Table 1, application, testing and the study of neutron dosimetry systems and individual detectors were accomplished for a number of "Benchmark Neutron Fields," research reactor "Test Regions," and PWR and BWR power reactor "In- and Ex-Vessel Surveillance Positions." The results of these dosimetry studies are being combined with available metallurgical data for use in the validation and calibration of 1) the established ASTM Standards and, 2) the currently applied reactor physics computational tools used for predicting PV steel toughness and embrittlement from test reactor and power reactor surveillance data.

With reference to Appendix A and Figures 1 and 2, a revised version of the Master Matrix Guide (0) was prepared by W. McElroy and successfully passed joint ASTM E10.05 and E10.02* ballot.(23) It is now being balloted at E10 and Society level. The Master Matrix Guide is now shown as one of a set of seventeen standards. In previous reports, fifteen standards were identified. A new ASTM Guide (IIF), "Correlation of ΔNDTT with Fluence," is shown in Figures 1 and 2. ASTM Subcommittee E10.02 will have the lead responsibility for its preparation, but the E10.05.06 Task Group on "Exposure Units" will assist E10.02 in its preparation. E10.02 has also taken the lead responsibility for the preparation of the ASTM Method (IIIE) "Temperature Monitors" of Figures 1 and 2.

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*ASTM E10 Subcommittee E10.02 on Behavior and Use of Nuclear Metallurgical Materials in Nuclear Systems and Subcommittee E10.05 on Nuclear Radiation Metrology (formerly, on Dosimetry).

A first draft was prepared by S. Anderson of the new ASTM Master Practice (IA), "Analysis and Interpretation of Nuclear Reactor Surveillance Results," which defines a consistent set of procedures for reactor pressure vessel fracture toughness and embrittlement predictions. It can be 1) referenced as an instrument of regulation and 2) used for the establishment of improved metallurgical data bases. A revised version of this standard is provided in Appendix B. The revision was made by S. Anderson and W. McElroy as a result of discussions associated with the balloting process within ASTM. It is noted that a new Appendix B, Figure 1, ASTM Standards for Surveillance of LWR Nuclear Reactor Pressure Vessels, has been prepared that shows 19 existing or proposed standards.

The E185-79 Standard, "Surveillance Tests for Nuclear Reactor Vessels," is now identified as IF and a new Standard IG, "Surveillance Tests for Nuclear Reactor Support Structures" has been identified and referenced in IA. The E184-79 standard, "Effects of High Energy Neutron Radiation of the Mechanical Properties of Metallic Materials," now serves as an interface standard between separate sets of standards related to the areas of (1) metallurgy surveillance and (2) dosimetry-physics-damage analysis surveillance.

Revised drafts of the ASTM Guide (IIA) for "Application of Neutron Spectrum Unfolding Methods" and the Guide (IIE) for "Benchmark Testing of Reactor Vessel Dosimetry" were prepared by F. Stallmann and E. McGarry/J. Grundl, respectively, and are in the process of review by program participants.

A revised draft of the ASTM Guide (IIC) for "Sensor Set Design," was prepared by G. Martin and a revised draft of the ASTM Method (IIIB) for "Solid State Track Recorder Monitors" was prepared by R. Gold, F. Ruddy, and J. Roberts for ASTM E10.05 Subcommittee ballot. The IIIB standard passed E10.05 ballot and is now being balloted at E10 and Society level.

A first draft of the ASTM Method (IIIC) for "Analysis of Helium Accumulation Fluence Monitors" was prepared by H. Farrar and B. Oliver and is in the process of review by program participants.

A first draft of the [illegible] was prepared by [illegible] in the process of review by program participants.

A first draft was prepared by [illegible]

[illegible]

[illegible]

[illegible]

IV. LWR PHYSICS - DOSIMETRY TESTING IN THE ORNL POOL CRITICAL ASSEMBLY PRESSURE VESSEL BENCHMARK FACILITY (ORNL-PCA)

[Variables Studied: 1) Plant dimensions - Core Edge to Surveillance to Vessel Wall to Support Structures Positions; 2) Core Power Distribution; 3) Reactor Physics Computations; 4) Selection of Neutron Exposure Units; 5) Neutron Spectral Effects; and 6) Dosimetry Measurements.]

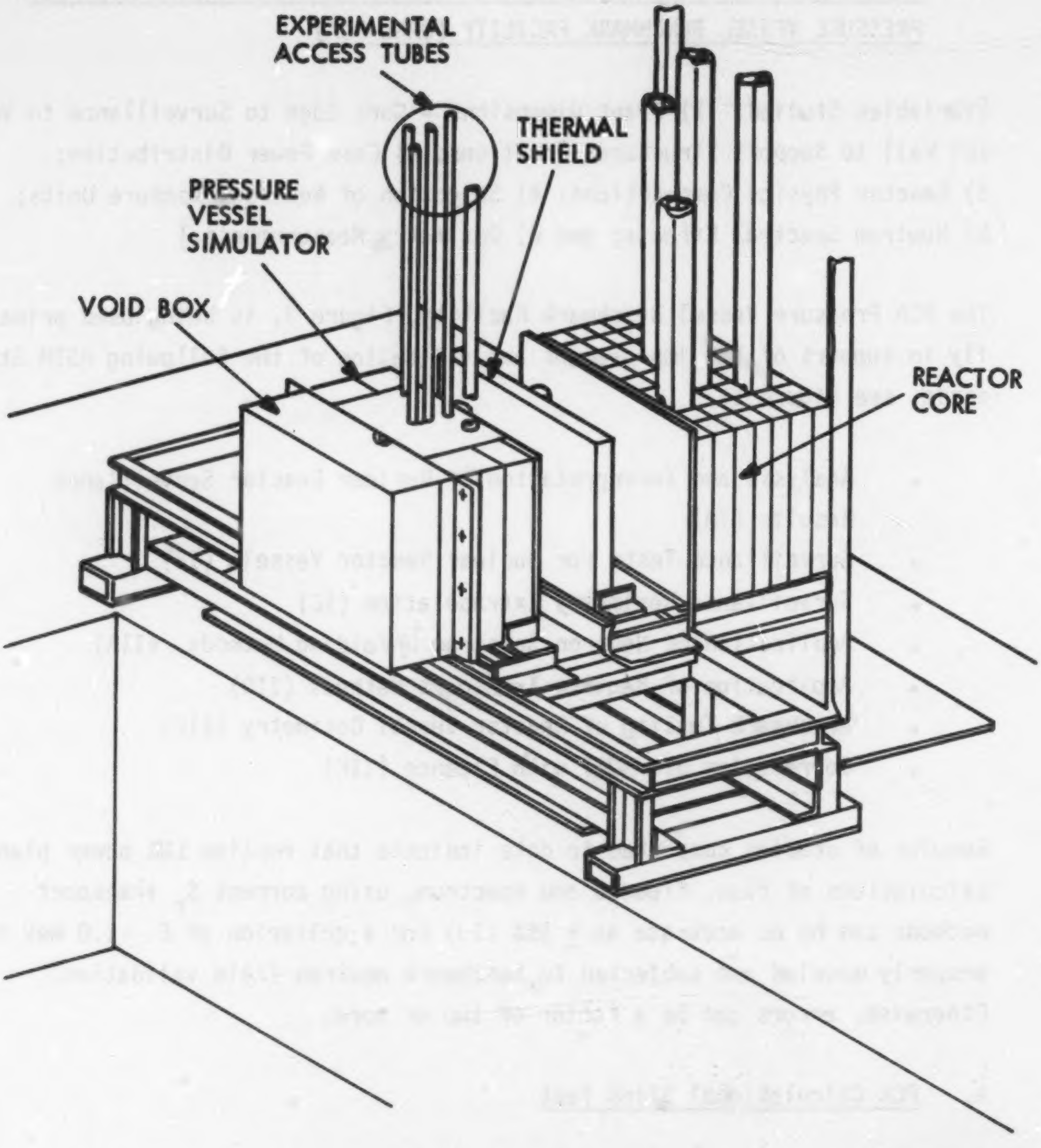
The PCA Pressure Vessel Benchmark Facility, Figure 3, is being used primarily in support of the improvement and validation of the following ASTM Standards, see Figure 1:

- Analysis and Interpretation of Nuclear Reactor Surveillance Results (IA)
- Surveillance Tests for Nuclear Reactor Vessels (IB)
- Surveillance Dosimetry Extrapolation (IC)
- Application of Neutron Spectrum Unfolding Methods, (IIA)
- Application of Neutron Transport Methods (IID)
- Benchmark Testing of Reactor Vessel Dosimetry (IIE)
- Correlation of NDTT with Fluence (IIF)

Results of studies completed to date indicate that routine LWR power plant calculations of flux, fluence and spectrum, using current S_n transport methods can be as accurate as ± 15% (1σ) for a criterion of $E > 1.0$ MeV if properly modeled and subjected to benchmark neutron field validation. Otherwise, errors can be a factor of two or more.

A. PCA Calculational Blind Test

The purpose of the PCA Calculational Blind Test is to determine how accurately transport theory calculations can predict neutron dosimetry measurements in a benchmark field which simulates a steel-water configuration of commercial light water reactors. A large number of dosimetry experiments were performed in the PCA-PV configuration which are used for comparison


EXPERIMENTAL ACCESS TUBES
THERMAL SHIELD
PRESSURE VESSEL SIMULATOR
VOID BOX
REACTOR CORE

HEDL 7804-028

FIGURE 3. Pressure Vessel Wall Mockup Schematic of Two Equivalent Facilities Constructed at ORNL. The high-flux version at ORR (PSF) will include damage exposure of metallurgical test specimens; the low-flux version near a low-power critical assembly (PCA) will focus on active and passive dosimetry measurements.

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with the calculations. To date, measurements of $^{237}Np(n,f)$, $^{103}Rh(n,n')$, $^{115}In(n,n')$, $^{238}U(n,f)$, $^{58}Ni(n,p)$, and $^{27}Al(n,\alpha)$ reaction rates are available at experimental locations A1, A3, A4, A5 and A6 in the PCA 12/13 configuration, and A4, A5, and A6 in the PCA 8/7 configuration, Figures 3 and 4.

To obtain the highest possible accuracy, all measurements (fission chambers and foils) were referenced to standard fission fields at NBS and CEN/SCK, Mol, Belgium. Active neutron and gamma spectroscopy and SSTR measurements have also been performed. Results of the neutron transport calculations of the PCA have been received from ten laboratories and vendors. The majority of them used a two-dimensional x-y geometry with some form of leakage correction. Two participants also submitted results from one-dimensional calculations and one participant used cylindrical geometry for the calculation. One Monte Carlo calculation was also submitted.

Compared with the measurements, most calculations showed a trend towards underestimating the fluxes by 5-25% and more so (50% and greater) for the cylindrical geometry. Only the one-dimensional calculations tended to overestimate the fluxes. Figure 5 shows an example of such a comparison in terms of the calculational to experimental (C/E) ratios for the $^{58}Ni(n,p)$ reaction at different locations in the PCA 12/13 configuration. This graph shows that not only are the calculations generally lower than the experimental values, but that the C/E ratios decrease also with increasing distance from the core.

If the same trend were found in the calculation of commercial power reactors, the extrapolation from surveillance measurements to the interior of the pressure vessel could lead to other large, nonconservative errors in flux and damage estimates. There is the possibility that leakage corrections may be responsible for the discrepancies. This and other possible explanations for the differences between calculated and measured results are being investigated.

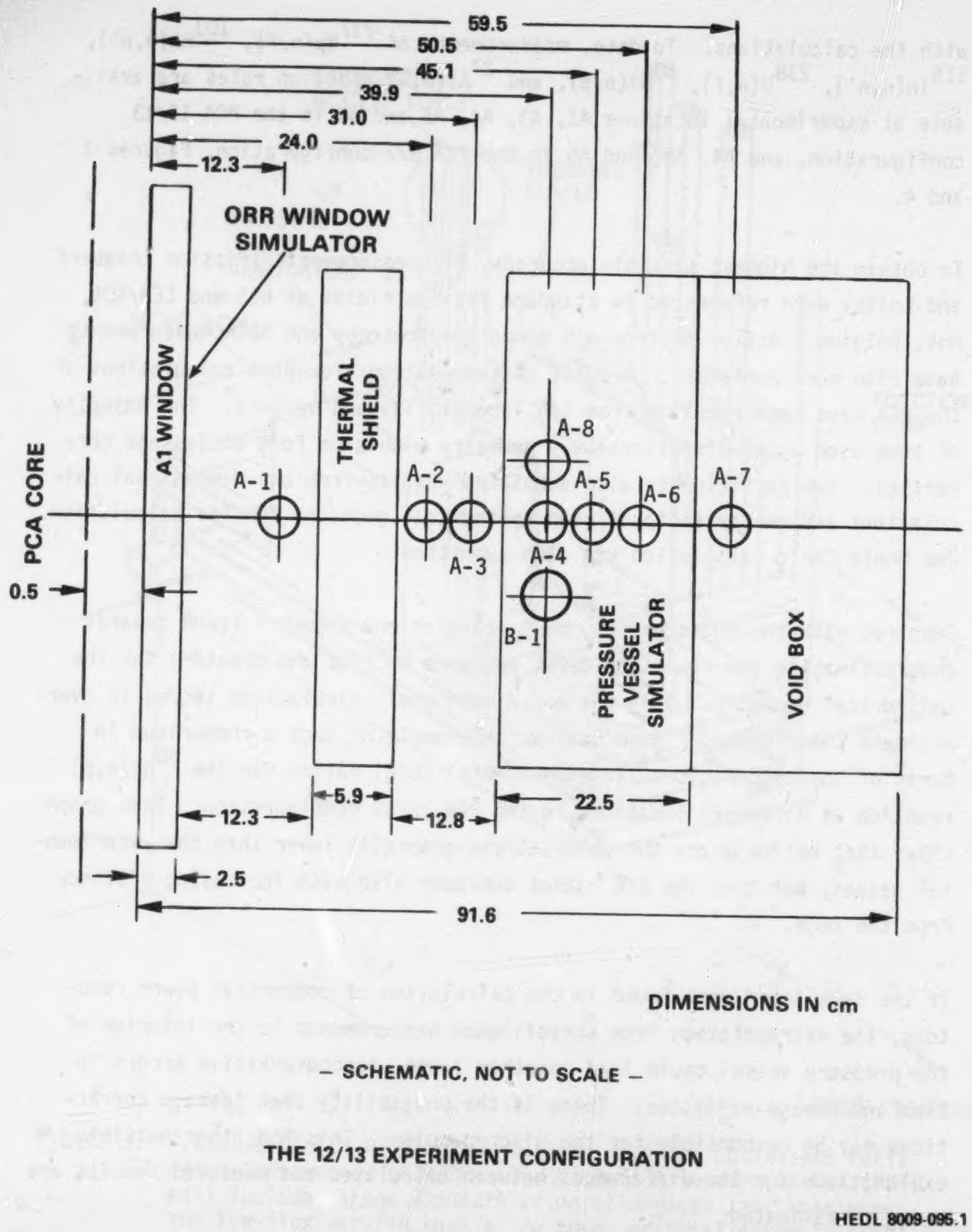
59.5
50.5
45.1
39.9
31.0
24.0
12.3
ORR WINDOW SIMULATOR
A1 WINDOW
THERMAL SHIELD
PCA CORE
A-1
A-2
A-3
A-4
A-5
A-6
A-7
A-8
B-1
PRESSURE VESSEL SIMULATOR
VOID BOX
0.5
12.3
5.9
12.8
22.5
2.5
91.6
DIMENSIONS IN cm
— SCHEMATIC, NOT TO SCALE —
THE 12/13 EXPERIMENT CONFIGURATION
HEDL 8009-095.1

FIGURE 4. Experimental Configuration: Locations A1, A2, A3, A4, A5, A6, A7, A8 and B1 in the PCA 12/13 Configuration.

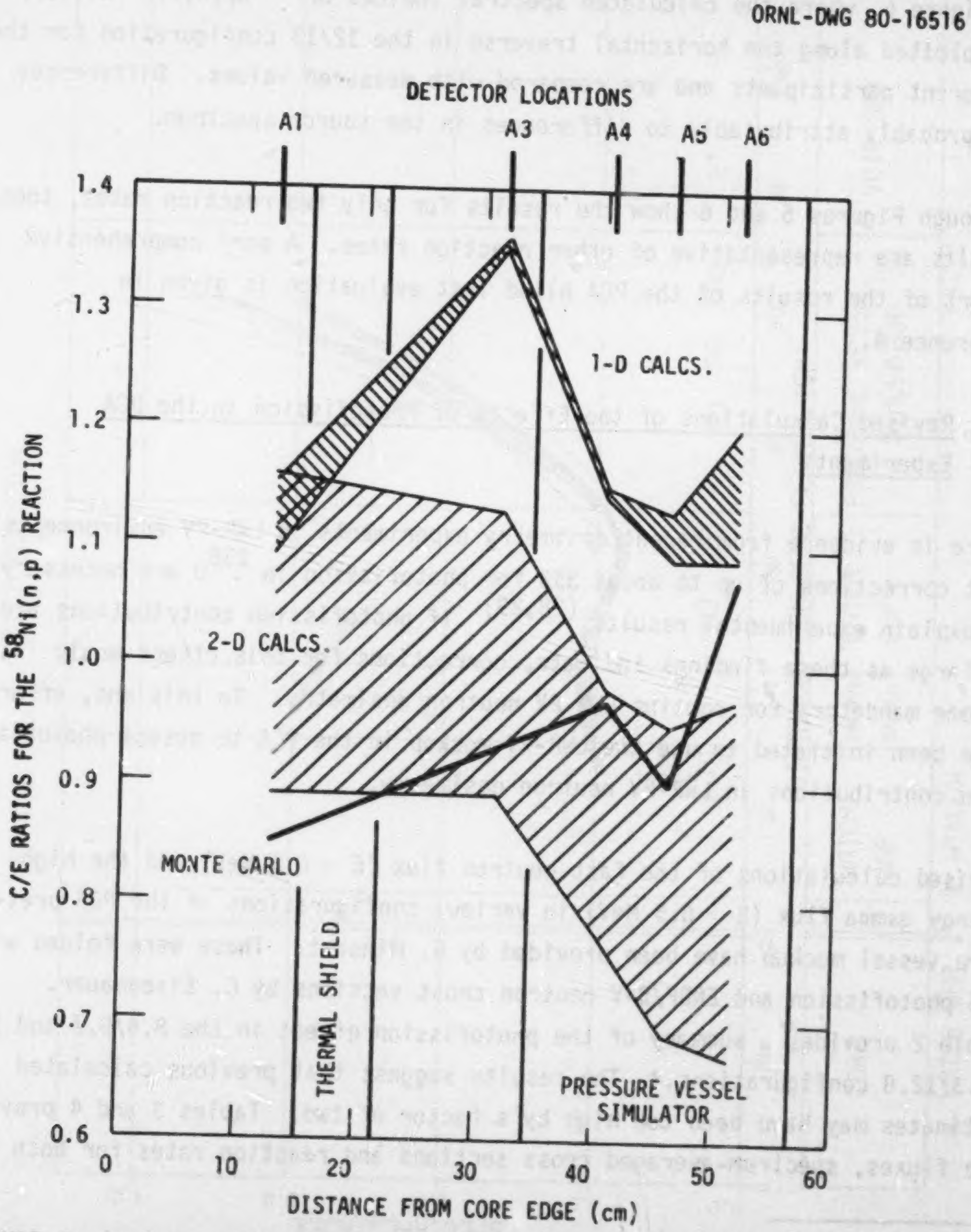

FIGURE 5. Ratios of Calculated to Experimental Reaction Rates for the ^{58}Ni(n,p) Reaction at Various Points in the PCA 12/13 Configuration. Cylindrical data not shown.

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The spectral shape represented by spectral indices, the ratio between two reaction rates, seems to be fairly accurately calculated. This can be seen in Figure 6, where the calculated spectral indices of ^{237}Np(n,f)/^{58}Ni(n,p) are plotted along the horizontal traverse in the 12/13 configuration for the different participants and are compared with measured values. Differences are probably attributable to differences in the source spectrum.

Although Figures 5 and 6 show the results for only two reaction rates, these results are representative of other reaction rates. A more comprehensive report of the results of the PCA Blind Test evaluation is given in Reference 4.

B. Revised Calculations of the Effects of Photofission in the PCA Experiments

There is evidence from recent dosimetry experiments in LWR-PV environments that corrections of up to about 35% for photofission in ^{238}U are necessary to explain experimental results.(28,29) If photofission contributions are as large as these findings indicate, corrections for this effect would become mandatory for routine LWR-PV neutron dosimetry. To this end, efforts have been initiated to use the LWR-PV mockup in the PCA to assess photofission contributions in LWR-PV neutron dosimetry.

Revised calculations of the fast neutron flux (E > 0.8 MeV) and the high-energy gamma flux (E > 6.5 MeV) in various configurations of the PCA pressure vessel mockup have been provided by G. Minsart. These were folded with NBS photofission and ENDF/B-V neutron cross sections by C. Eisenhauer. Table 2 provides a summary of the photofission effect in the 8.4/6.7 and the 12.3/12.8 configurations.* The results suggest that previous calculated estimates may have been too high by a factor of two. Tables 3 and 4 provide the fluxes, spectrum-averaged cross sections and reaction rates for both

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*The 8.4/6.7 and 12/3/12/8 are the actual dimensions (cm) of the 8/7 and 12/13 configurations, respectively.

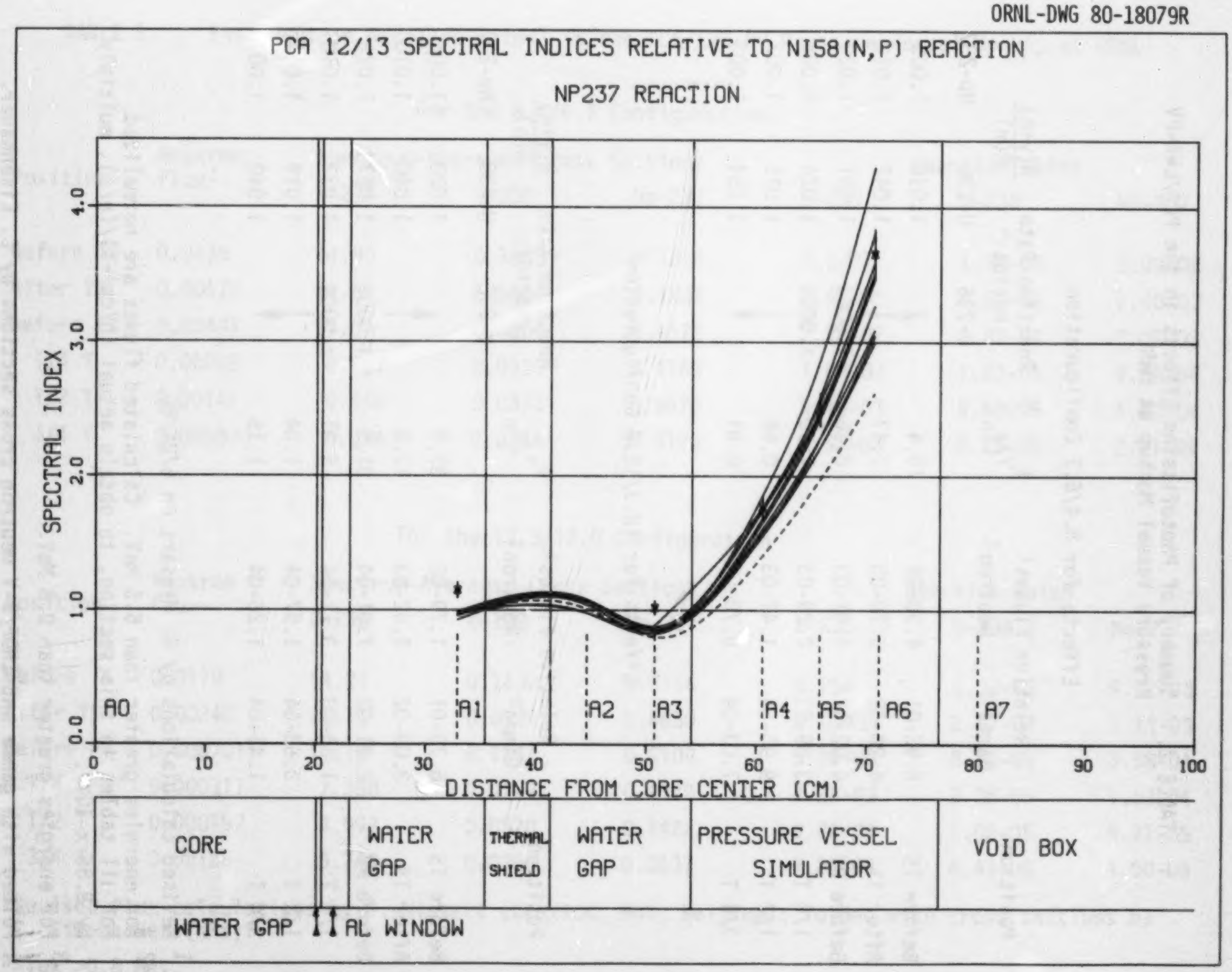

FIGURE 6. $^{237}Np(n,f)/^{58}Ni(n,p)$ Plot for PCA 12/13 Configuration.

TABLE 2. Summary of Photofission Effects in the PCA/LWR-PV Pressure Vessel Mockup at ORNL

Effects for 8.4/6.7 Configuration

Position	Relative Fluxes[1]		ϕ_γ/ϕ_n	Reaction Rate Ratios:[4] $\frac{R(\gamma+n)}{R(n)}$		
	Gamma[2]	Neutron[3]		U-235	U-238	Np-237
Before TS	8.92-01	4.38-02	20.4	↑	1.016	1.005
After TS	4.26-02	5.70-03	7.47		1.047	1.018
Before PV	4.02-02	4.81-03	8.36	1.0003	1.051	1.020
1/4 T	3.89-03	2.29-03	1.70	±.0003	1.024	1.006
1/2 T	9.95-04	1.47-03	0.68		1.015	1.003
3/4 T	7.03-04	8.70-04	0.81	↓	1.054	1.008

Effects for 12.3/12.8 Configuration

Position	Relative Fluxes[1]		ϕ_γ/ϕ_n	Reaction Rate Ratios: $\frac{R(\gamma+n)}{R(n)}$		
	Gamma[2]	Neutron[3]		U-235	U-238	Np-237
Before TS	6.70-01	1.79-02	37.4	↑	1.020	1.006
After TS	3.09-02	2.42-03	12.8		1.040	1.016
Before PV	2.21-02	7.01-04	31.5	1.0006	1.067	1.028
1/4 T	1.65-03	3.11-04	5.31	±.0004	1.037	1.010
1/2 T	3.63-04	1.97-04	1.84		1.019	1.004
3/4 T	1.45-04	1.26-04	1.15	↓	1.040	1.007

[1] Revised calculations by G. Minsart on 6/24/80.

[2] For energies greater than 6.5 MeV. Calculated fluxes are normalized. For all tables in this section, to obtain actual $(n/cm^2\text{-}s)/(n/s)$, multiply by 8.86×10^{-4}.

[3] For energies greater than 0.8 MeV.

[4] Folded with gamma and ENDF/B-V neutron cross sections by C. Eisenhauer. Reaction rates are per nucleus per 1 core neutron per second.

TABLE 3. Fast Neutron Reaction Rates[1] in the PCA/LWR-PV Pressure Vessel Mockup at ORNL

For the 8.4/6.7 Configuration

Position	Neutron Flux[2]	Spectrum-Averaged Cross Sections			Reaction Rates		
		U-235	U-238	Np-237	U-235	U-238	Np-237
Before TS	0.0438	14.90	0.1003	0.4772	6.5-01	1.39-03	2.09-02
After TS	0.00570	14.82	0.0801	0.4202	8.45-02	4.57-04	2.40-03
Before PV	0.00481	15.01	0.0804	0.4173	7.22-02	3.87-04	2.01-03
1/4 T	0.00229	8.207	0.0535	0.4163	1.88-02	1.23-04	9.53-04
1/2 T	0.00147	6.450	0.0373	0.3670	9.48-03	5.48-05	5.40-04
3/4 T	0.000870	6.334	0.0254	0.3120	5.51-03	2.21-05	2.71-04

For the 12.3/12.8 Configuration

Position	Neutron Flux[2]	Spectrum-Averaged Cross Sections			Reaction Rates		
		U-235	U-238	Np-237	U-235	U-238	Np-237
Before TS	0.0179	14.21	0.1130	0.5156	0.254	2.02-03	9.23-03
After TS	0.00242	14.11	0.0939	0.4650	0.0341	2.27-04	1.13-03
Before PV	0.000701	14.18	0.1141	0.5109	9.94-03	8.00-05	3.58-04
1/4 T	0.000311	7.350	0.0727	0.4929	2.28-03	2.26-05	1.53-04
1/2 T	0.000197	4.997	0.0520	0.4422	9.84-04	1.02-05	8.71-05
3/4 T	0.00126	5.743	0.0350	0.3637	7.24-04	4.41-06	4.58-05

[1] Revised flux calculations by G. Minsart (CEN/SCK, Mol, Belgium); folded with cross sections by C. Eisenhauer (NBS).

[2] Flux >0.8 MeV.

TABLE 4. Photofission Reaction Rates[1] in the PCA/LWR-PV Pressure Vessel Mockup at ORNL

For the 8.4/6.7 Configuration

Position	Gamma Flux[2]	Spectrum-Averaged Cross Sections			Reaction Rates		
		U-235	U-238	Np-237	U-235	U-238	Np-237
Before TS	0.8923	1.06-04	7.88-05	1.27-04	9.46-05	7.03-05	1.13-04
After TS	0.0426	8.27-04	5.00-04	9.98-04	3.52-05	2.13-05	4.25-05
Before PV	0.0402	8.13-04	4.93-04	9.76-04	3.27-05	1.98-05	3.92-05
1/4 T	0.00389	1.26-03	7.76-04	1.46-03	4.90-06	2.98-02	5.68-06
1/2 T	0.000995	1.37-03	8.39-04	1.58-03	1.36-06	8.35-07	1.57-06
3/4 T	0.000703	2.82-03	1.70-03	3.17-03	1.98-06	1.195-06	2.23-06

For the 12.3/12.8 Configuration

Position	Gamma Flux[2]	Spectrum-Averaged Cross Sections			Reaction Rates		
		U-235	U-238	NP-237	U-235	U-238	Np-237
Before TS	0.6698	7.28-05	5.88-05	8.64-05	4.88-05	3.94-05	5.79-05
After TS	0.0309	4.71-04	2.94-04	5.73-04	1.46-05	9.09-06	1.77-05
Before PV	0.0221	3.78-04	2.42-04	4.52-04	8.36-06	5.35-06	9.99-06
1/4 T	0.00165	8.00-04	5.00-04	9.25-04	1.32-06	8.25-07	1.53-06
1/2 T	0.000363	8.68-04	5.43-04	1.00-03	3.15-07	1.97-07	3.63-07
3/4 T	0.000145	1.97-03	1.20-03	2.23-03	2.86-07	1.74-07	3.23-07

[1] Revised flux calculations by G. Minsart (CEN/SCK, Mol, Belgium); folded with cross sections by C. Eisenhauer (NBS).

[2] Flux >6.5 MeV.

neutrons and gammas for the U-235, U-238 and Np-237 fissionable isotopes. Table 2 also shows truncated gamma and neutron fluxes and their ratios. The latter is generally a good indicator of the relative photofission effect. In these tables, the letters TS designate the thermal shield; PV, the pressure vessel; and 1/4-, 1/2-, and 3/4-T the quarter, half and three-quarter thicknesses within the pressure vessel simulator.

The largest photofission corrections are required for U-238 fission and they now appear to be less than 7% (rather than the 15% of previous estimates) for all Table 2 locations. Corrections for fission in Np-237 are only about one third as large, and those for U-235 are completely negligible. However, these results are still very preliminary, since the accuracy of the calculated corrections is very much in question.

Most of the photofission gammas are believed to arise from thermal-neutron capture in iron. In the PCA mockups, calculations indicate that the photofission gammas are generated locally in surrounding or nearby iron. For example, when gamma rays from the core are increased by a factor of 3, the gamma ray flux in the pressure vessel simulator increases by only about 20%. This is in contrast to PWR and BWR configurations where the gammas causing photofission in front of the pressure vessel are believed to be generated in the thermal shield. Therefore, calculated gamma fluxes are sensitive to the calculated neutron flux. The accuracy of calculated thermal fluxes can be assessed by comparison with some measured ^{235}U(n,f) reaction rates. Reaction rates calculated by G. Minsart are shown in Table 5 and are compared to some experimental data measured by F. Ruddy, R. Gold, and J. Roberts with bare and cadmium-covered solid state track recorders. Two experimental fission chamber data points by A. Fabry and E. McGarry are also shown. Since large discrepancies are observed between theory and measurement for the thermal neutron component, these calculated photofission contributions cannot be regarded as reliable.

Additional calculations have been performed, but these do not shed any light on the thermal neutron discrepancies. In this regard, Table 6 compares

TABLE 5

CALCULATED AND EXPERIMENTAL REACTION RATE AND Cd RATIOS FOR ^{235}U(n,f) IN THE PCA/LWR-PV MOCKUP

Position	Reaction Rate Ratio[1] (Cd-Covered)	Reaction Rate Ratio[1] (Bare)	Cd Ratio Calculated	Cd Ratio Experimental
8.4/6.7 Configuration				
1/4T	1.00	1.37	2.54	1.86
1/2T	1.05	1.24	1.33	1.12
3/4T	1.30	6.52	6.00	1.20
12.3/12.8 Configuration				
Before TS			127.	130[2]
Before PV			428.	125[2]
1/4T	1.10		6.10	
1/2T	0.93	1.20	1.61	1.25
3/4T	1.20	6.02	5.50	1.10

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[1] G. Minsart's revised calculations are divided by experimental results obtained by HEDL with bare and cadmium-covered solid state track recorders. The ratio is normalized at the 8.4/6.7 1/4T position.

[2] Experimental data obtained with Mol fission chambers.

TABLE 6. Comparison of Minsart's (MOL) Revised Cadmium Ratios with those Calculated by Siman-Tov (ORNL) for the 12/13 Configuration

Position	Siman Tov (12.3/12.8) Cadmium Ratios	G. Minsart (12.3/12.8) Cadmium Ratios
Before TS	77.0	127.
After TS	200.	160.
Before PV	480.	428.
1/4 T	4.25	6.10
1/2 T	1.65	1.61
3/4 T	25.0	5.50

G. Minsart's calculated cadmium ratios with the results of one-dimensional calculations by Siman Tov (ORNL). Except at the 3/4 T position where Siman Tov's results are significantly higher, agreement between the two calculations is good. Finally, the presence of Cd around the fission chambers may also produce a significant number of photofission gammas. Rough calculations by C. Eisenhauer indicate a 4% effect on $^{238}U(n,f)$ compared to estimates of up to 7% for gammas generated in iron.

Additional efforts are clearly required to adequately resolve photofission effects in LWR-PV neutron dosimetry. The forthcoming experimental schedule for the PCA reflects this need and, in particular, includes absolute gamma-ray spectrometry so that calculations can be directly tested.

V. LWR Steel Metallurgical Testing in the Pressure Vessel Benchmark Facility (ORR-PSF)

[Variables Studied: 1) Steel Chemical Composition and Microstructure; 2) Steel Irradiation Temperature; 3) Selection of Neutron Exposure Units; 4) Dosimetry Measurements; and 5) Neutron Spectra and Dose Rate Effects.]

The LWR Metallurgical Pressure Vessel Benchmark Facility (ORR-PSF) is being used primarily in support of the improvement and validation of the following ASTM Standards, see Figure 1:

- Analysis and Interpretation of Nuclear Reactor Surveillance Results (IA)
- Surveillance Tests for Nuclear Reactor Vessels (IB)
- Surveillance Dosimetry Extrapolation (IC)
- Displaced Atom (DPA) Exposure Unit (ID)
- Damage Correlation (IE)
- Application of Neutron Spectrum Unfolding Methods (IIA)
- Sensor Set Design (IIC)
- Correlation of ΔNDTT with Fluence (IIF)
- Five Methods, IIIA, IIIB, IIIC, IIID, and IIIE

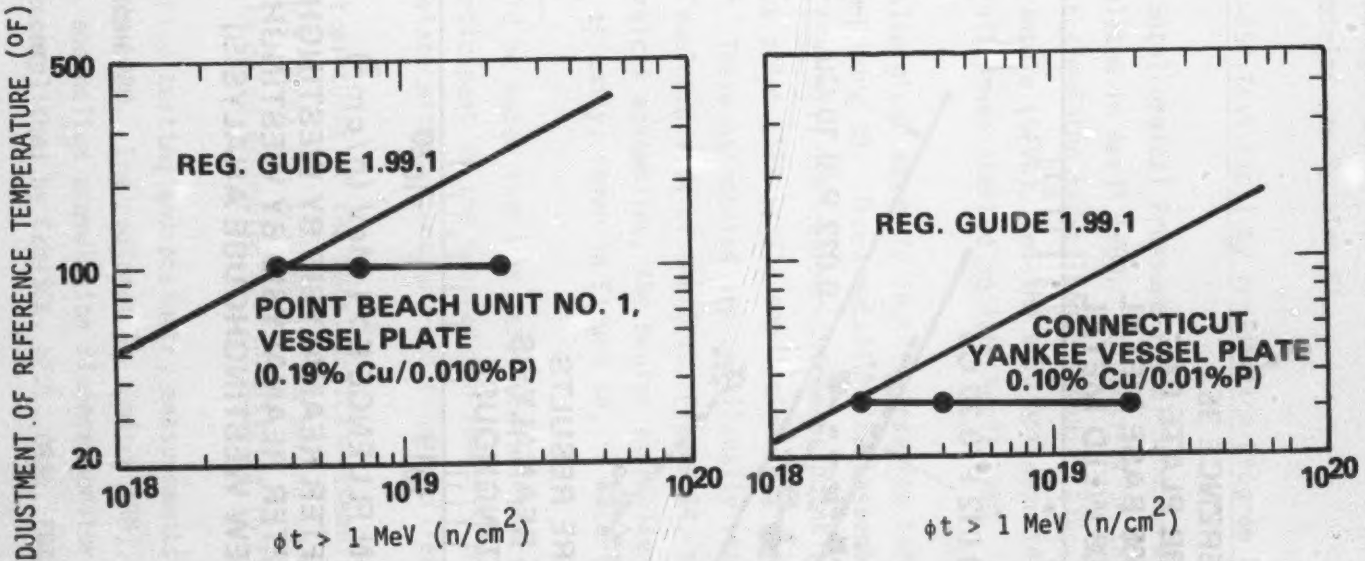
There are a number of metallurgical programs and studies that have been established to determine the fracture toughness and Charpy properties of irradiated materials as a function of chemistry, microstructure, and irradiation conditions.(3,7-9,11,13-16,30-33,35-47) The ORR-PSF multilaboratory metallurgical-dosimetry improvement program(1,2,11,25-27), is expected to provide key irradiation effects data, under well controlled conditions, to help in 1) the validation and calibration of exposure units and values and 2) the analysis and correlation of property change data obtained from this and other programs. Summary information on recent results of power and test reactor data development and testing and the status of the ORR-PSF program work is presented in Sections V-A, B, and C.

A. Power and Test Reactor Metallurgical Data Development and Testing

Recent analyses of PWR surveillance capsule and test reactor data indicate that long-term LWR power plant surveillance capsule and short-term test reactor ($\sim 288^{o}C$ irradiation temperature) neutron-induced property change data for steel (base metal, heat affected zone and weld metal) can show significantly different fluence dependencies.(2,3,14-16,30,31,33,36,46) For instance, for low-flux surveillance capsule irradiated materials, the neutron induced damage may increase at a rate per unit fluence similar to that of high flux test reactor irradiated materials, up to some level of exposure which appears to be a function of chemistry and microstructure. At exposures above this level, the rate of embrittlement is much reduced and it appears that the embrittlement saturates. As an example, Figure 7, the Point Beach Unit No. 1 and Connecticut Yankee vessel plate, weld material, and the A302B correlation materials do indicate apparent saturation at exposures up to about 2×10^{19} n/cm^2 (E > 1 MeV).[14,36] On the other hand, Figure 8, the Point Beach Unit No. 2 surveillance capsules V, T, and R vessel plate, weld material, and A533B correlation material results do not show such saturation.(31) Also, in Figure 8, the Point Beach Unit No. 2 A533B results (dashed curve for 0.14% Cu) appear to correlate well with those (two solid curves for 0.10 and 0.20% Cu) obtained from high flux test reactor studies.(7,31,35)

Based on the above results as well as those from other studies,(6,9,37) it appears that the constant A and the slope N, of the equation $\Delta NDTT = A(\phi t)^N$, are both functions of chemistry. With reference to the preparation of the ASTM IE Correlation Practice, Figure 1, and in addition to the inclusion of a thermal annealing term to account for possible saturation effects, it is concluded that such equations may have to include N as a function of Cu, P, etc.

In summary and in consideration of the application of ORR-PSF experimental data, test reacto and surveillance capsule irradiation effects data should not be combined to predict PV steel fracture toughness and embrittlement as



NOTE: THE VESSEL WELDPLATE AND A302B CORRELATION MATERIALS, NOT SHOWN HERE, ALSO SHOW APPARENT SATURATION.

HEDL 8009-054.2

FIGURE 7. Point Beach Unit No. 1 and Connecticut Yankee Trend Curve Results.

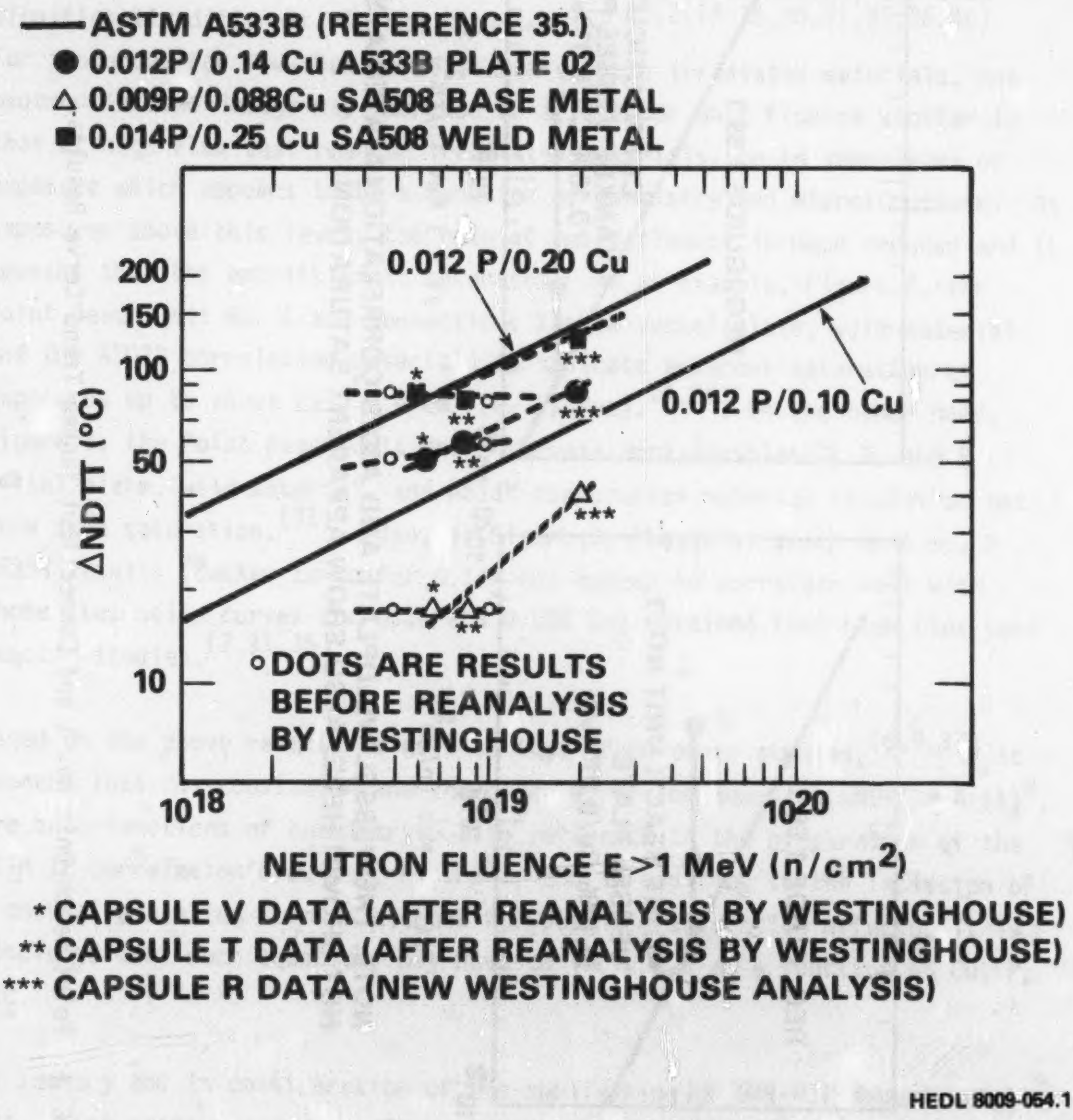


FIGURE 8. Point Beach Unit No. 2 Trend Curve Results.

a function of neutron exposure without having 1) more precisely defined and representative data bases, 2) a better understanding of the mechanisms causing damage, and 3) tested and verified exposure data and physical correlation models; see Section VI.

B. ORR-PSF Test Facility Irradiation Program Status

The LWR Metallurgical Pressure Vessel Benchmark Facility, shown in Figure 9, was completed in April 1980. It consists of three irradiation capsules--the first simulated surveillance capsule (SSC-1), the simulated pressure vessel capsule (SPVC), and the void box capsule (VBC). Capsule irradiation at ORR full power started in late April 1980.

The metallurgical assembly that is inserted into the surveillance capsule is shown in Figure 10. Three metallurgical assemblies located at the inside surface (SPV-0T), the quarter-thickness point (SPV-1/4T), and the half-thickness point (SPV-1/2T), are loaded into the simulated pressure vessel capsule. These assemblies are individually shown in Figures 11, 12, and 13. An exploded view of the SPVC, Figure 14, shows the positions of the metallurgical assemblies, the heater assemblies, and the cooler assemblies. A fifth assembly, shown in Figure 15, is inserted into the void box capsule.

It should be noted that the metallurgical specimens are all labeled, so that each participant in the program can identify the location of their specimens to correlate with temperature and flux profiles.(2)

As-built pictures of the metallurgical assemblies, designed at ORNL, are shown in Figures 16, 17, and 18. A second surveillance capsule, SSC-2, is currently being assembled for irradiation in FY-81.

Pre-startup testing and dosimetry measurements were completed in late 1979 and early 1980. The radiometric monitor (RM), solid state track recorder (SSTR), and helium acumulation fluence monitor (HAFM) dosimetry analysis is being accomplished by CEN/SCK, HEDL, ORNL, EG&G ORTEC, Harwell, and RI staff members.

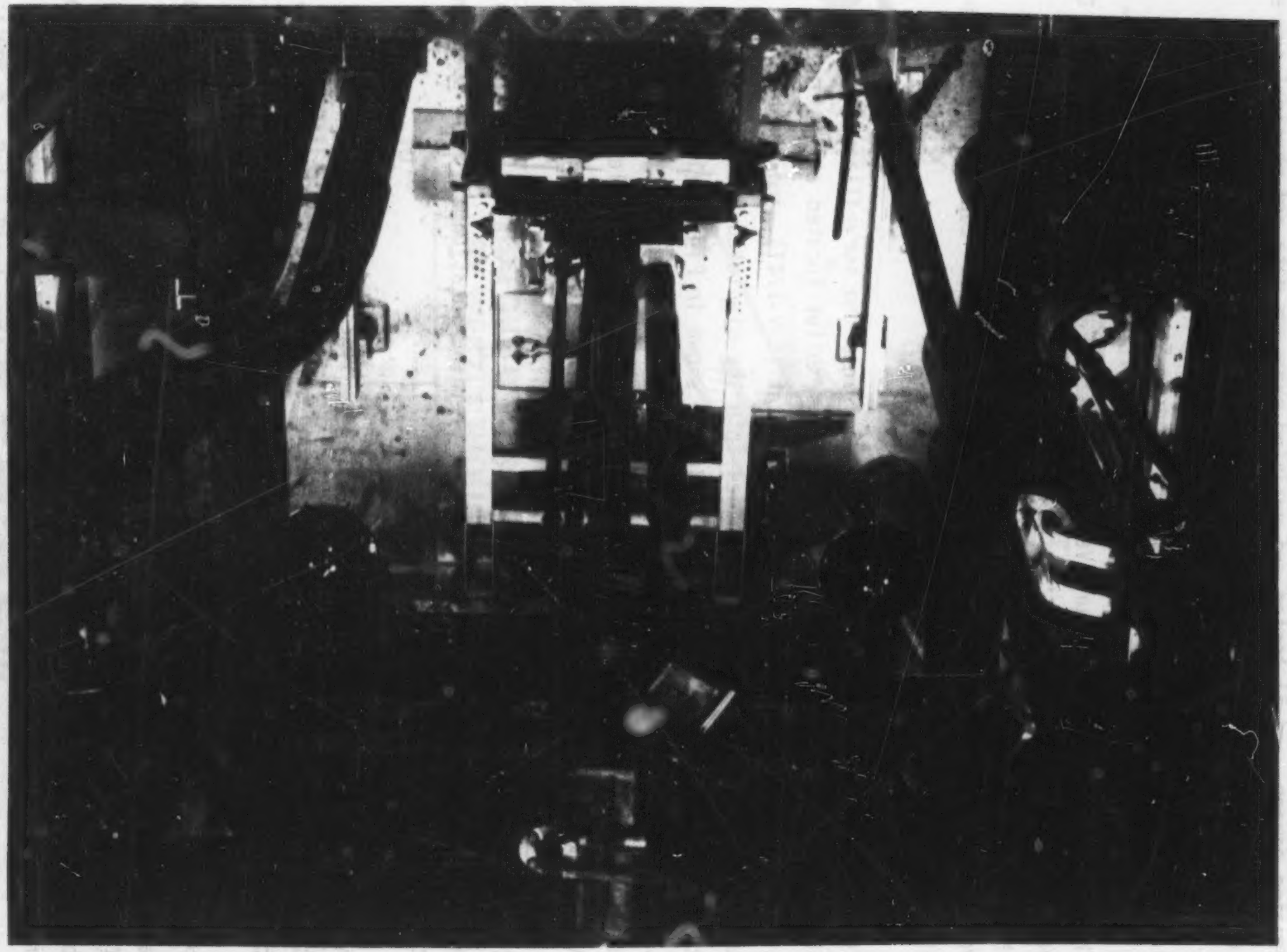


FIGURE 9. The LWR Metallurgical Pressure Vessel Benchmark Facility. Neg 8010270-19

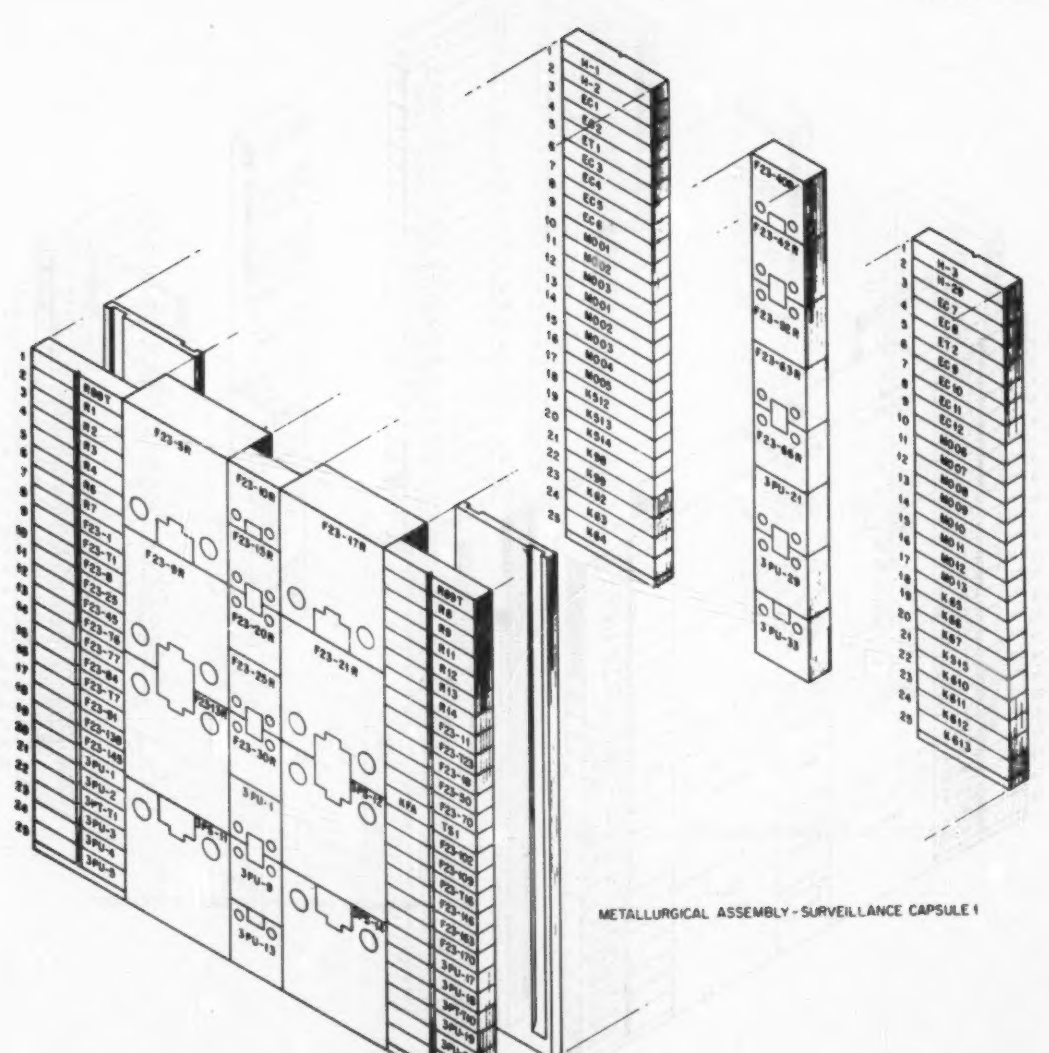

ORNL-DWG 79-17783R
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
ROOT
R1
R2
R3
R4
R6
R7
F23-1
F23-T1
F23-8
F23-25
F23-45
F23-T6
F23-77
F23-84
F23-T7
F23-91
F23-138
F23-148
3PU-1
3PU-2
3PT-T1
3PU-3
3PU-4
3PU-5
F23-5R
F23-9R
F23-13R
3PS-11
F23-10R
F23-15R
F23-20R
F23-25R
F23-30R
3PU-1
3PU-9
3PU-13
F23-17R
F23-21R
3PS-12
KFA
ROOT
R8
R9
R11
R12
R13
R14
F23-11
F23-123
F23-18
F23-50
F23-70
TS1
F23-102
F23-109
F23-T16
F23-116
F23-163
F23-170
3PU-17
3PU-18
3PT-T10
3PU-19
3PU-20
3PU-21
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
H-1
H-2
EC1
EC2
ET1
EC3
EC4
EC5
EC6
MO01
MO02
MO03
MO01
MO02
MO03
MO04
MO05
KS12
KS13
KS14
K60
K61
K62
K63
K64
F23-40R
F23-42R
F23-52R
F23-63R
F23-66R
3PU-21
3PU-29
3PU-33
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
H-3
H-20
EC7
EC8
ET2
EC9
EC10
EC11
EC12
MO06
MO07
MO08
MO09
MO10
MO11
MO12
MO13
K65
K66
K67
KS15
K610
K611
K612
K613
METALLURGICAL ASSEMBLY - SURVEILLANCE CAPSULE 1

FIGURE 10. The Metallurgical Assembly Located in the Surveillance Capsule.
Neg 8010270-10

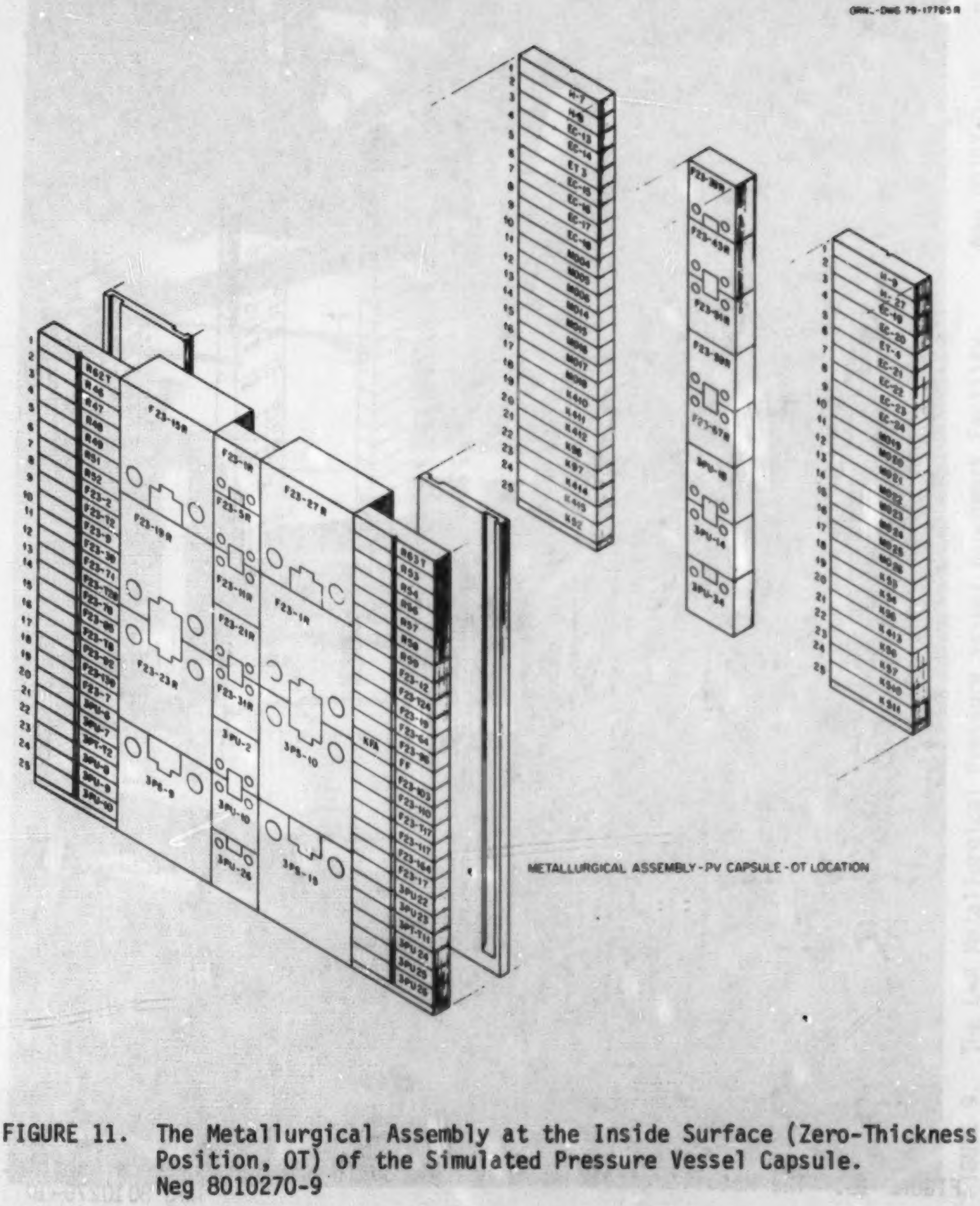

ORNL-DWG 79-17765R
METALLURGICAL ASSEMBLY-PV CAPSULE-OT LOCATION
R62T
R46
R47
R51
R52
F23-2
F23-72
F23-9
F23-30
F23-71
F23-7
3PU-6
3PU-7
3PT-T2
3PU-8
3PU-9
3PU-10
F23-15R
F23-19R
F23-23R
3PS-9
F23-1R
F23-5R
F23-11R
F23-21R
F23-31R
3PU-2
3PU-10
3PU-26
F23-27R
F23-1R
3PS-10
3PS-15
KFA
R63T
R53
R54
R56
R57
F23-12
F23-124
F23-19
F23-64
FF
F23-103
F23-110
F23-117
F23-164
F23-17
3PU22
3PU23
3PT-T11
3PU24
3PU25
3PU26
H-7
EC-13
EC-14
ET3
EC-15
EC-16
EC-17
MD04
MD05
MD06
MD14
MD15
MD17
K410
K411
K412
K97
K414
K415
K52
F23-43R
F23-67R
3PU-14
3PU-34
H-27
EC-20
ET-4
EC-21
EC-22
EC-23
EC-24
MD20
MD21
MD22
MD23
MD24
MD25
K53
K54
K413

FIGURE 11. The Metallurgical Assembly at the Inside Surface (Zero-Thickness Position, OT) of the Simulated Pressure Vessel Capsule.
Neg 8010270-9

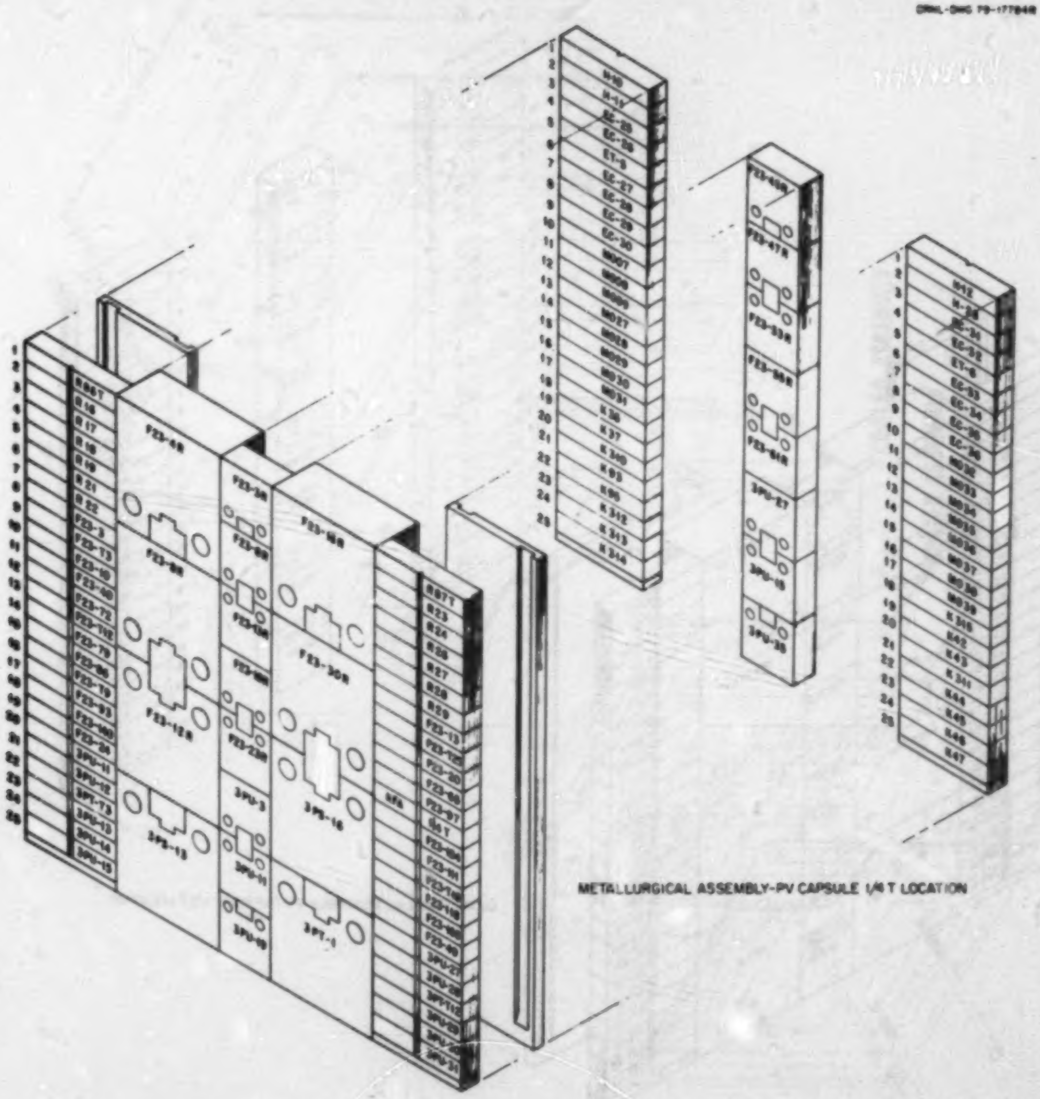
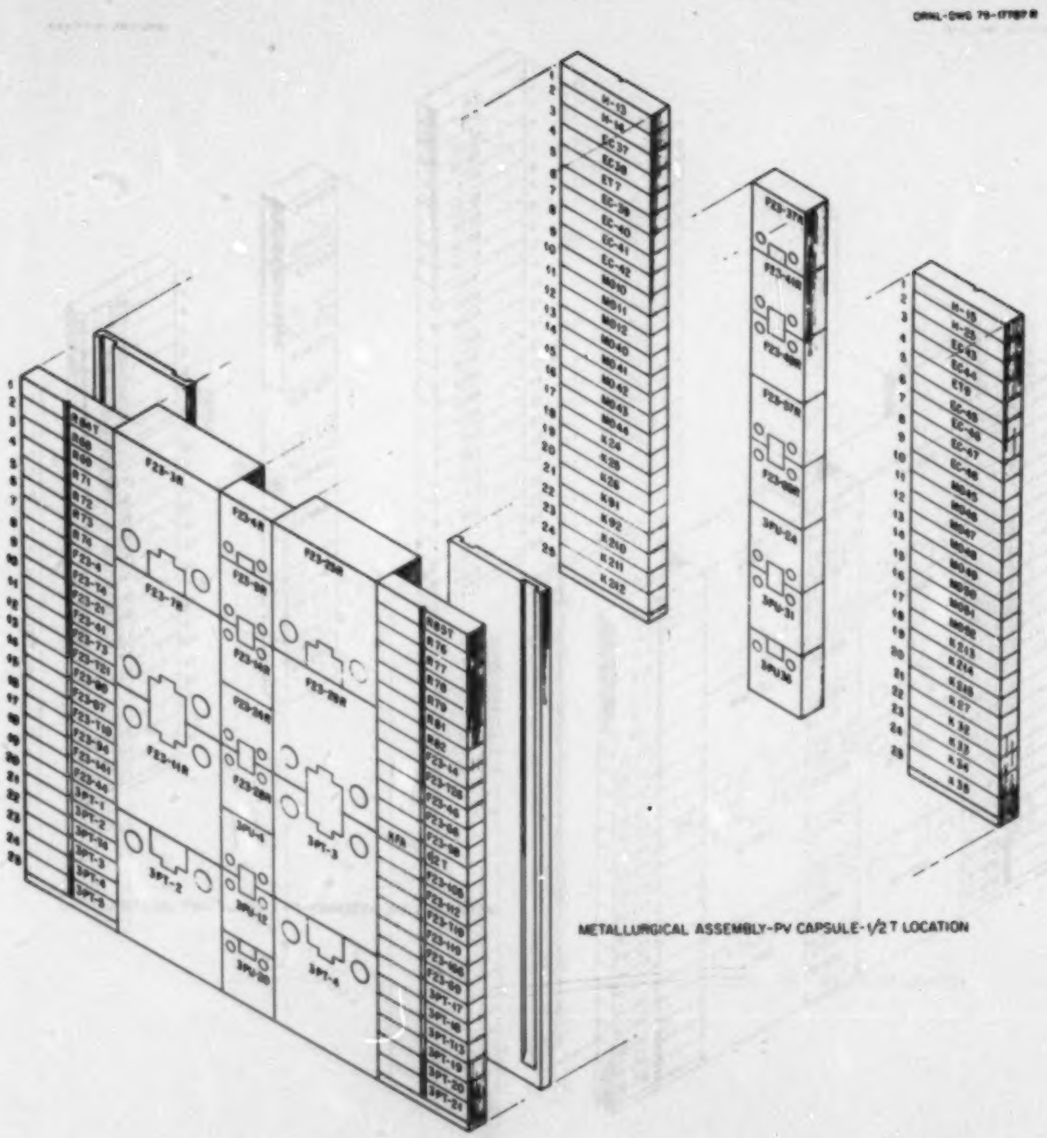

ORNL-DWG 79-17784R
METALLURGICAL ASSEMBLY-PV CAPSULE 1/4 T LOCATION

FIGURE 12. The Metallurgical Assembly Located at the Quarter-Thickness (1/4T) Position in the Simulated Pressure Vessel Capsule. Neg 8010270-8



ORNL-DWG 79-17787 R
METALLURGICAL ASSEMBLY-PV CAPSULE-1/2 T LOCATION
H-13
H-14
EC 37
EC 38
ET 7
EC-39
EC-40
EC-41
EC-42
MO10
MO11
MO12
MO40
MO41
MO42
MO43
MO44
K24
K25
K26
K91
K92
K210
K211
K212
H-15
H-25
EC43
EC44
ET8
EC-45
EC-46
EC-47
EC-48
K213
K214
K27
F23-3R
F23-4R
F23-7R
F23-11R
F23-25R
F23-29R
3PT-1
3PT-2
3PT-3
3PT-4
3PU-12
3PU-24
3PU-31
R71
R72
R73
R74
R76
R77
R78
R79
R81
R82
3PT-17
3PT-19
3PT-20
3PT-21

FIGURE 13. The Metallurgical Assembly Located at the Half-Thickness (1/2T) Position in the Simulated Pressure Vessel Capsule. Neg 8010270-7

ORNL DWG 79-10107

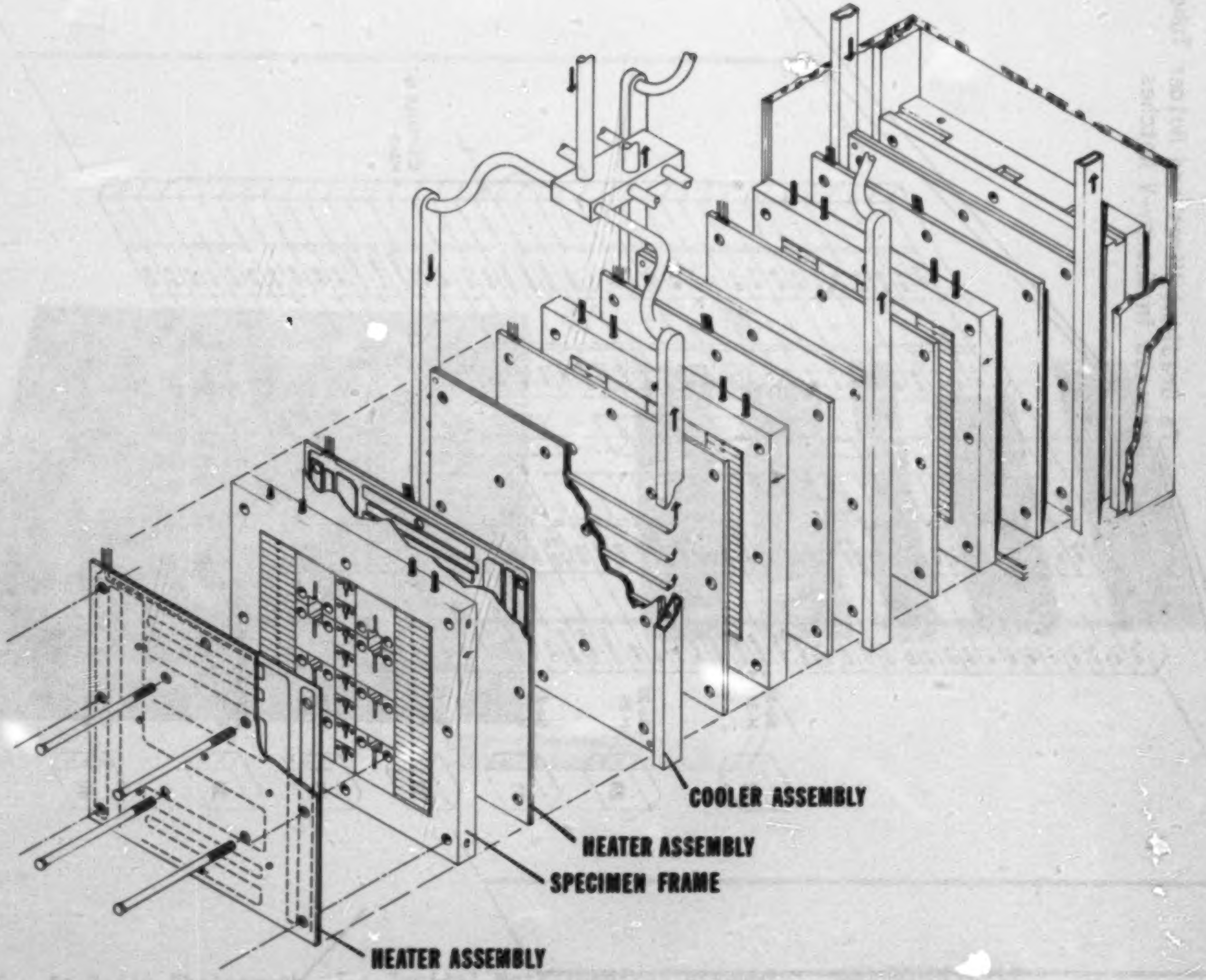

COOLER ASSEMBLY
HEATER ASSEMBLY
SPECIMEN FRAME
HEATER ASSEMBLY

FIGURE 14. Exploded View of the Simulated Pressure Vessel Capsule. Neg 8010270-15

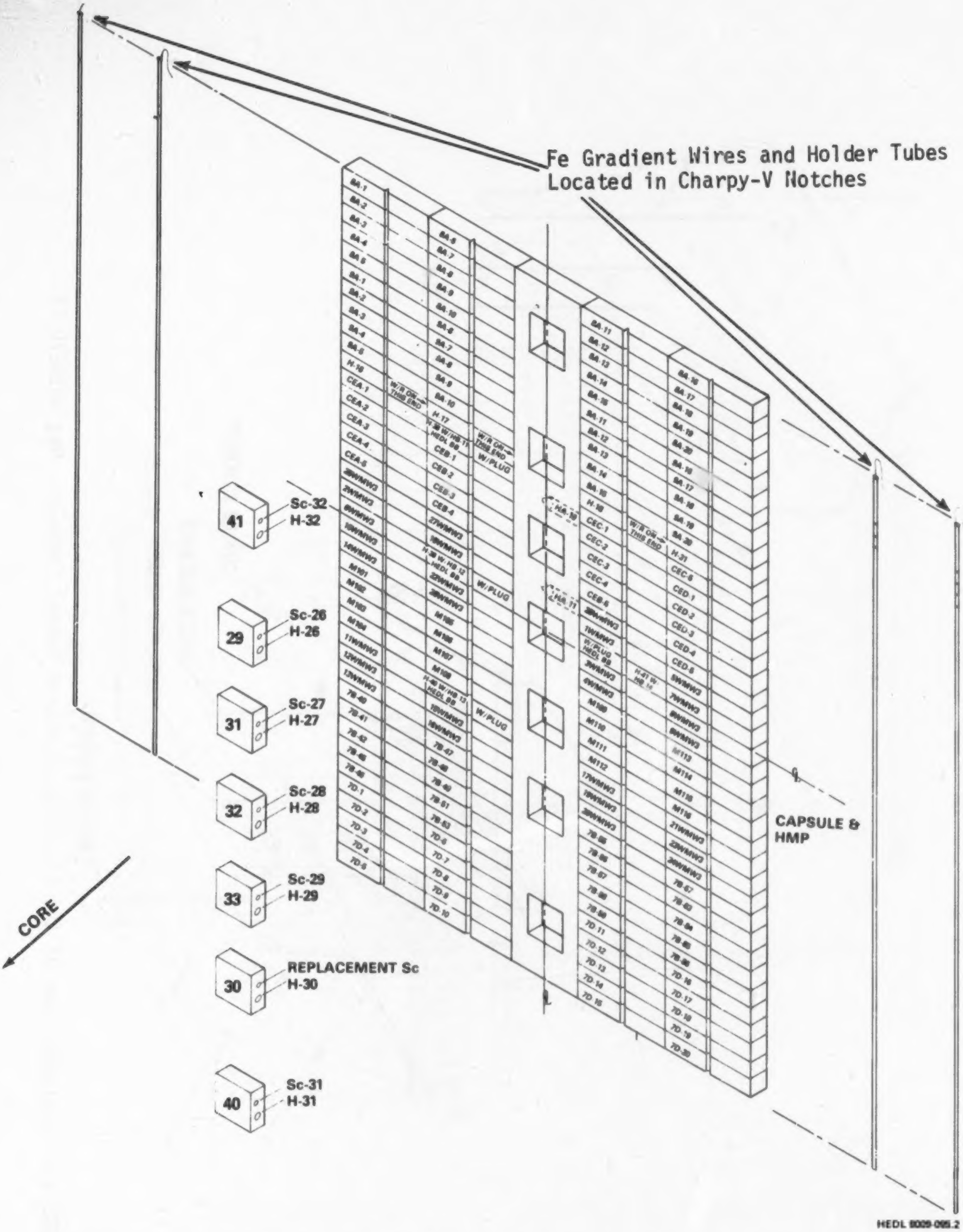

Fe Gradient Wires and Holder Tubes Located in Charpy-V Notches
CAPSULE & HMP
CORE
41
Sc-32
H-32
29
Sc-26
H-26
31
Sc-27
H-27
32
Sc-28
H-28
33
Sc-29
H-29
30
REPLACEMENT Sc
H-30
40
Sc-31
H-31
W/R ON THIS END
W/PLUG
HEDL 8009-095.2

FIGURE 15. The Metallurgical Assembly Contained in the Void Box Capsule (VBC).

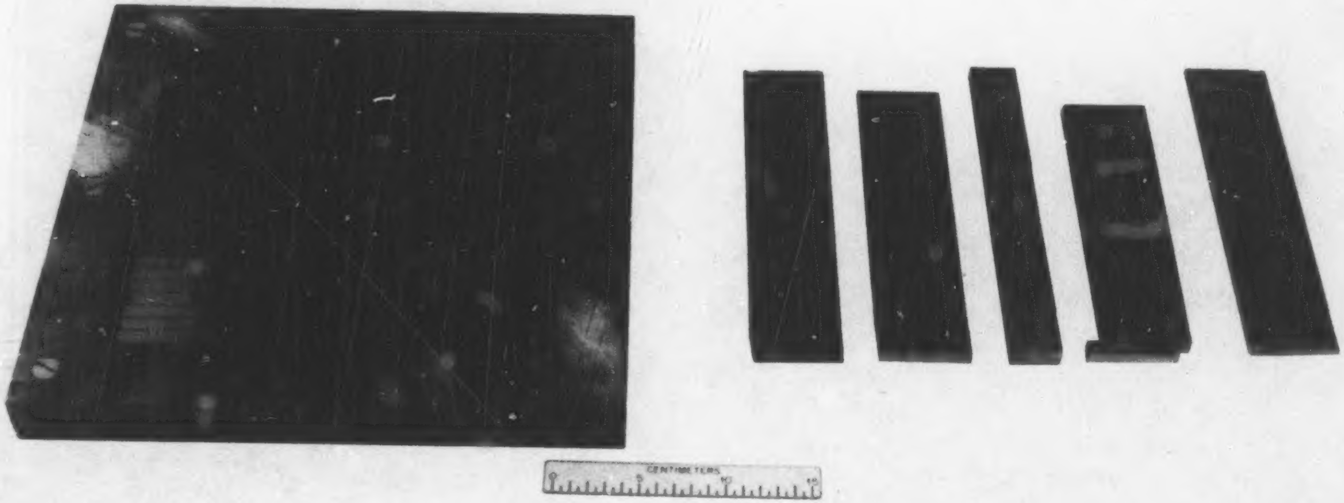

CENTIMETERS
0
5
10
15

FIGURE 16. As-Built Photograph of a Typical Metallurgical Assembly. Neg 8010270-17

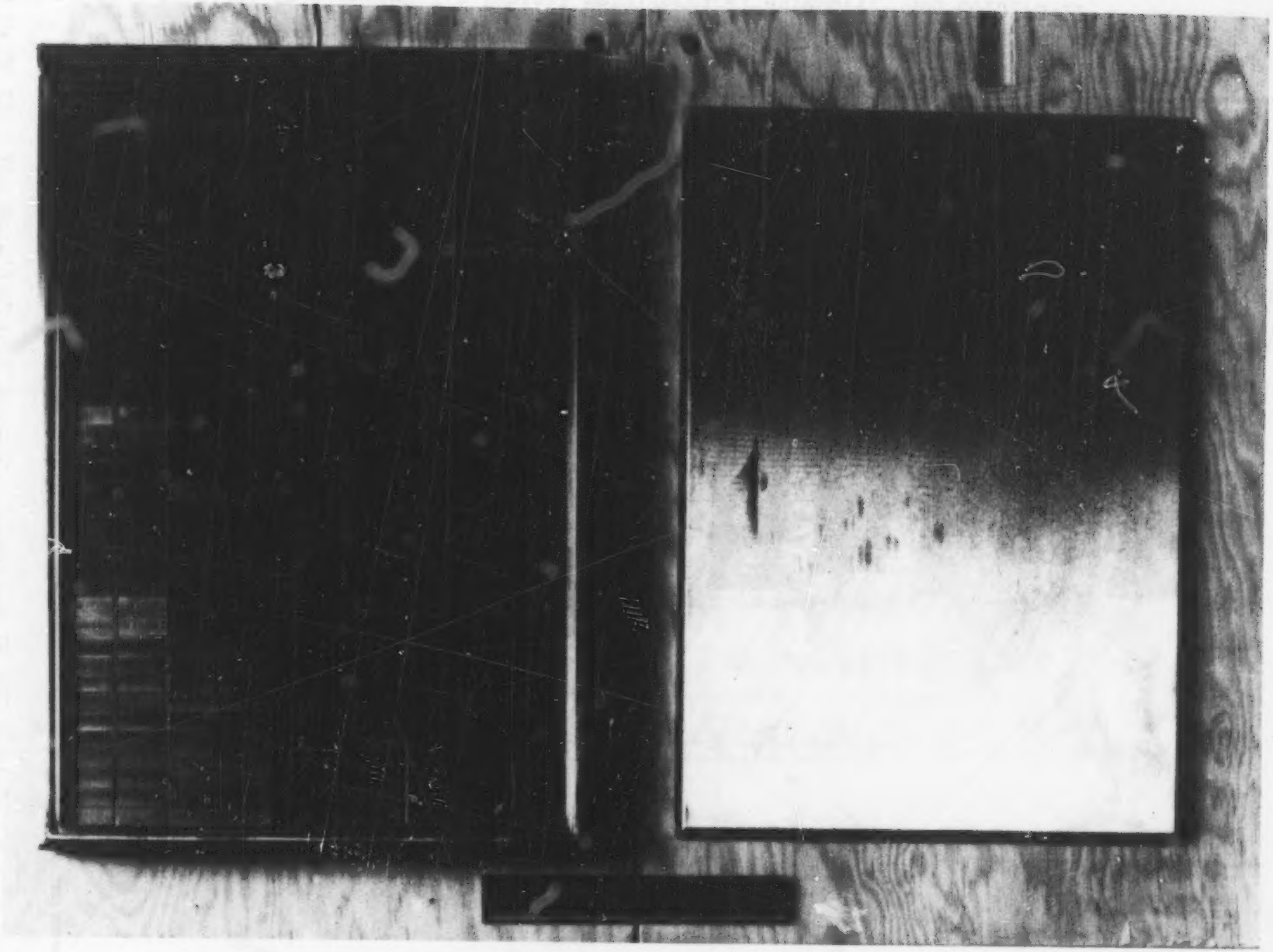


FIGURE 17. Rear View of the Void Box Metallurgical Assembly, Showing Iron Gradient Wires and Holder Tubes Positioned in Charpy Notches. Neg 8010270-18

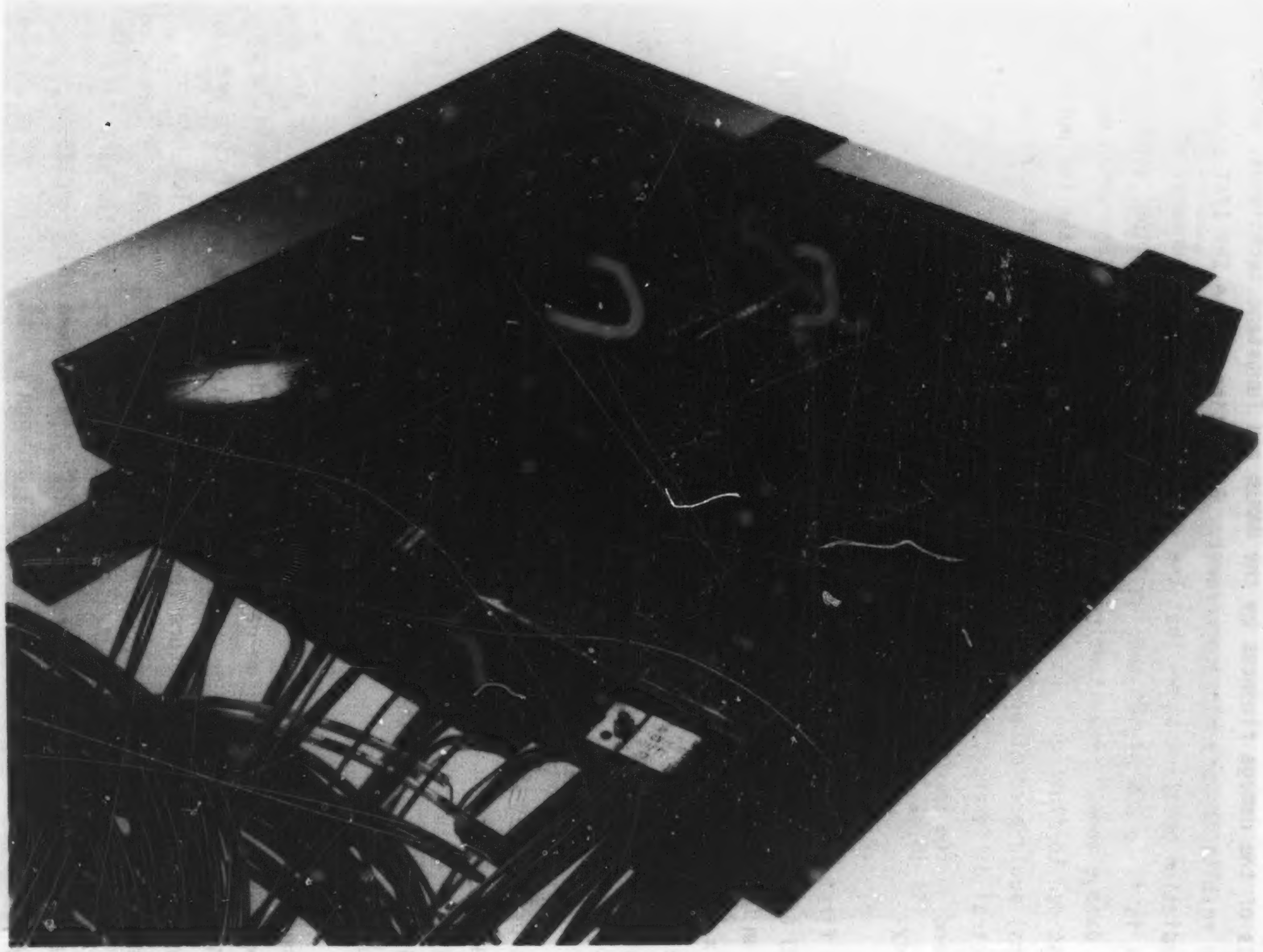


FIGURE 18. An Assembled Pressure Vessel Capsule, Showing Thermocouple Leads. Neg 8010270-21

A. Alberman, M. Benoist, and M. Thierry of CEN, Saclay, completed measurements of the damage fluences by the GAMIN and Tungstene techniques, which gave results in very close agreement. Evaluation of the SPV 1/4T steel irradiation position shows that the damage effectiveness per unit of nickel flux is 20 to 30% higher than at the SSC surveillance position, depending on the damage model. They concluded, therefore, that any effect which may exist due to flux level (a factor of 10 difference between the two positions) would predominate over the effect due to the 20 to 30% damaging spectral differences.(34) Certainly, the validity of this conclusion will depend on the chemistry and microstructure of the steels under study.(14-16,31,35-46)

The first surveillance capsule, SSC-1, was removed at 13:53 hours, June 23, 1980 after approximately 45 days of irradiation at 30 MW, and is presently in an ORNL hotcell awaiting decapsulation. The accumulated fluence of $\sim 2 \times 10^{19}$ n/cm^2 ($E > 1$ MeV) will correspond to the fluence received at the SPV-1/4 T assembly after 570 days of irradiation. It is anticipated that SSC-1 metallurgical and dosimetry capsules will be shipped to program participants by September 1980.

The process control panel, which monitors, controls, and records the temperature profiles in the SSC-1 and SPVC is shown in Figure 19. The VBC contains no temperature monitors. The temperature profile data is currently being processed for the SSC-1, which has already acquired the designed neutron fluence.

Temperature control is accomplished through the use of coolant water and a mixture of neon and helium gas, which provide decoupling of the metallurgical assemblies in the SPVC, and electrical heaters. The gas mixture and the coolant water set temperature for fixed heating conditions. The heaters, controlled by Variac settings under computer supervision, then are turned on or off to maintain the specified temperature profiles. Preliminary data analysis indicates that temperatures will lie between 281°C and 293°C, with the bulk of the data at approximately 288°C.

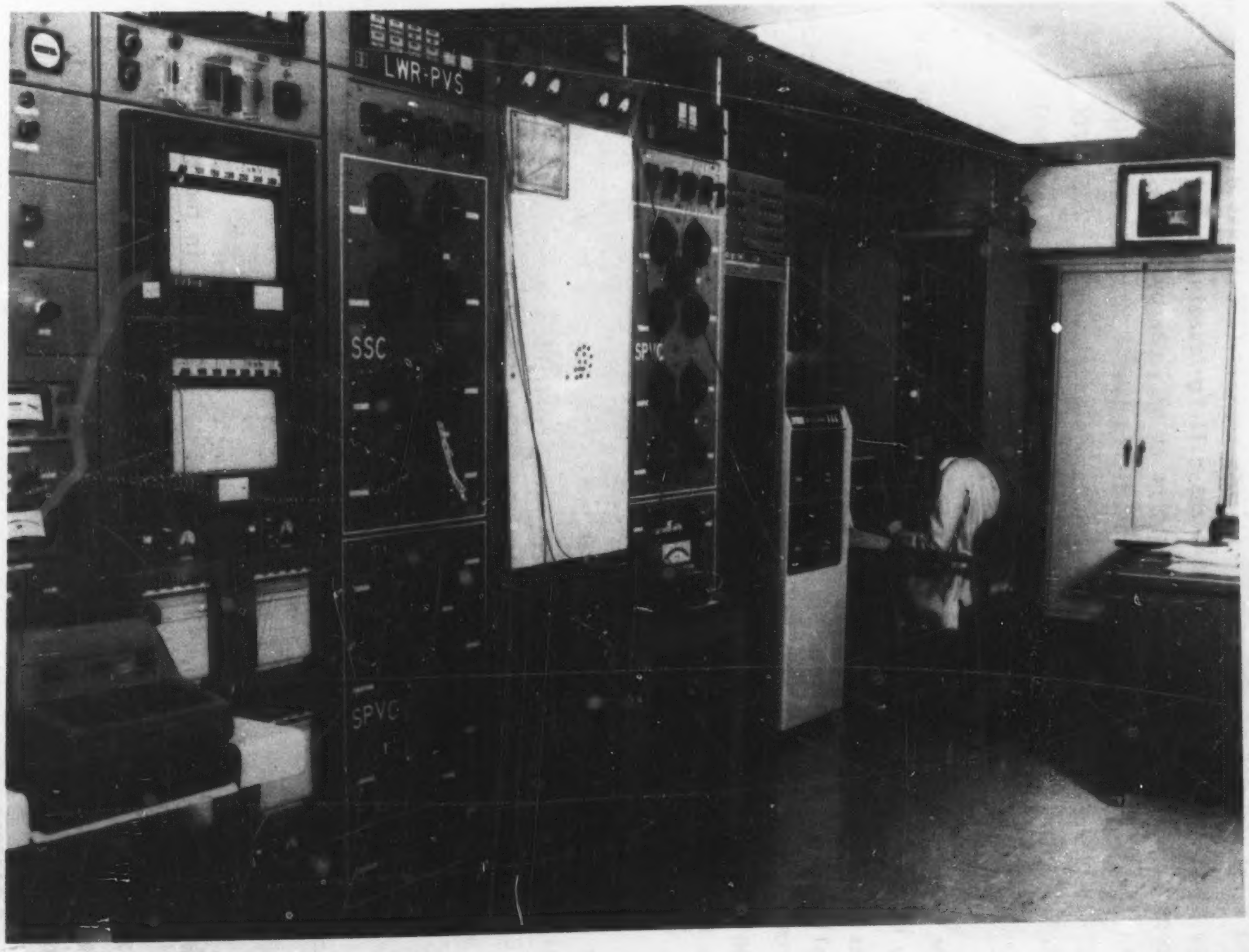

LWR-PVS
SSC
SPVC
SPVC

FIGURE 19. The Process Control Panel for the Pressure Vessel Benchmark Facility. Neg 8010270-20

C. ORR-PSF Metallurgical Specimen Test Program

The first ORR-PSF simulated surveillance capsule (SSC-1) was disassembled in September 1980. Present testing plans call for examination of Charpy and Compact Tension samples by J. Hawthorne at NRL, Charpy samples at KFA, and examination of Transmission Electron Microscopy (TEM), Hardness (H) and Cylindrical Compression (CC) samples by personnel at HEDL, UCSB, FCC, CEN/SCK, Harwell, and elsewhere. Physics and dosimetry analysis will be a joint effort among the participating laboratories; see Figure 21.

The preparation for testing of compression cylinders at HEDL has been done by K. Carlson. The tests will be done in an "MTS" hydraulic compression testing machine using a small sub-press shown in Figure 20. The MTS load cell will be used for recording the stress while the strain will be obtained from a linear variable transformer attached to the sub-press. Operation of the equipment is planned for temperatures in the range from room temperature up to approximately 260°C. The number and types of compression samples to be tested are identified in Tables 7 and 8.

In addition to its use to help test and verify physically based damage correlation models, the information obtained from the compression tests on the NRC/NRL A302B and A533B plate material (Tables 9 and 10) can be used to help plan the distribution of test temperatures for Charpy specimens. This will optimize the ability of NRL to obtain an accurate determination of the 30 ft-lb and 50 ft-lb charpy-energy temperatures from the limited number of Charpy samples, Table 9. Charpy V-notch impact data for unirradiated control specimens for the A302B and A533B reference plate materials have already been obtained by NRL.

The compression cylinder specimens are being irradiated in two types of holders 1) transverse drilled blocks (e.g., block 38 of Figure 21) and 2) Charpy-sized carriers containing nine cylinders each as shown in Figure 22. The latter type specimens cover the metallurgical specifications shown in Table 10 and are part of a cooperative investigation by University of

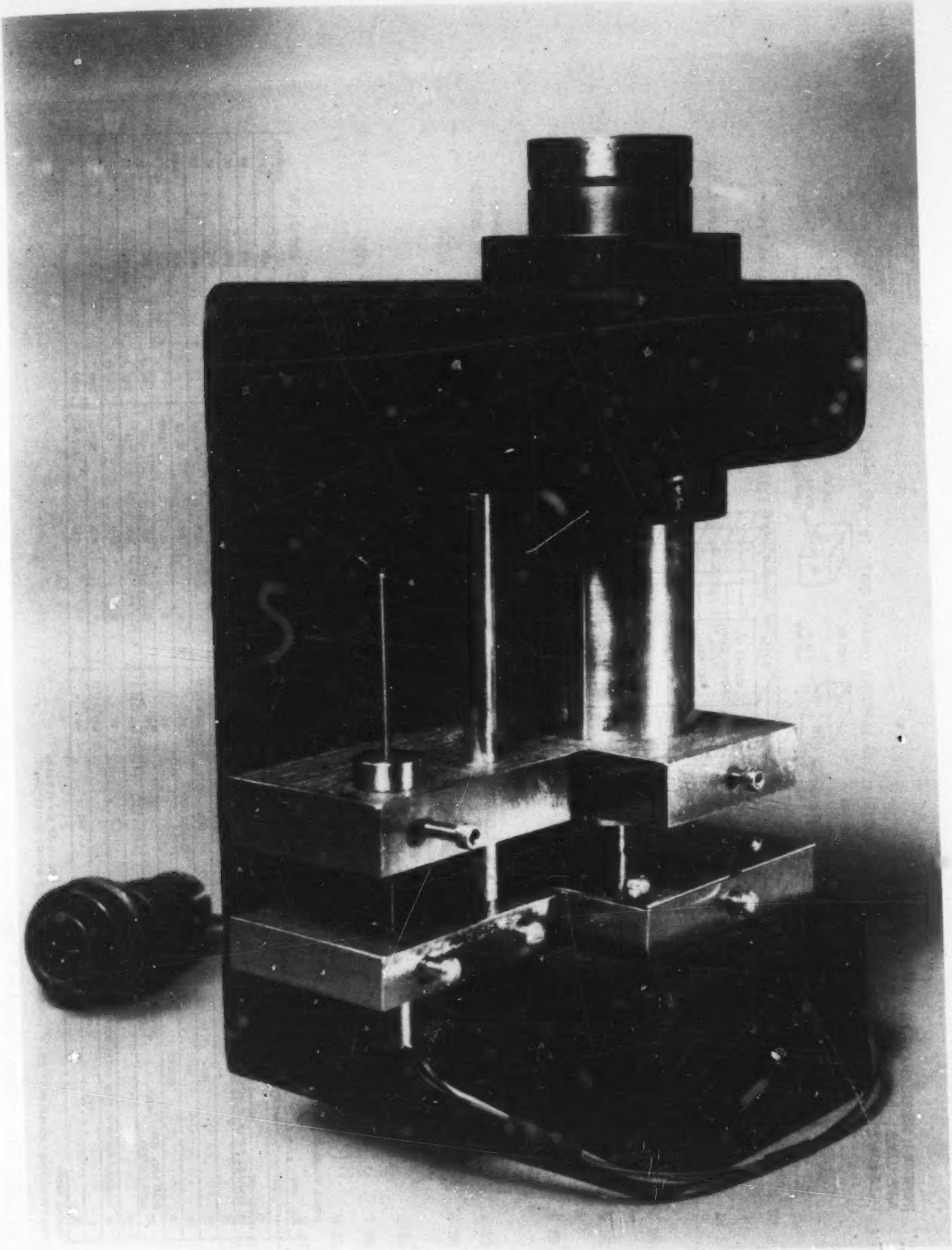
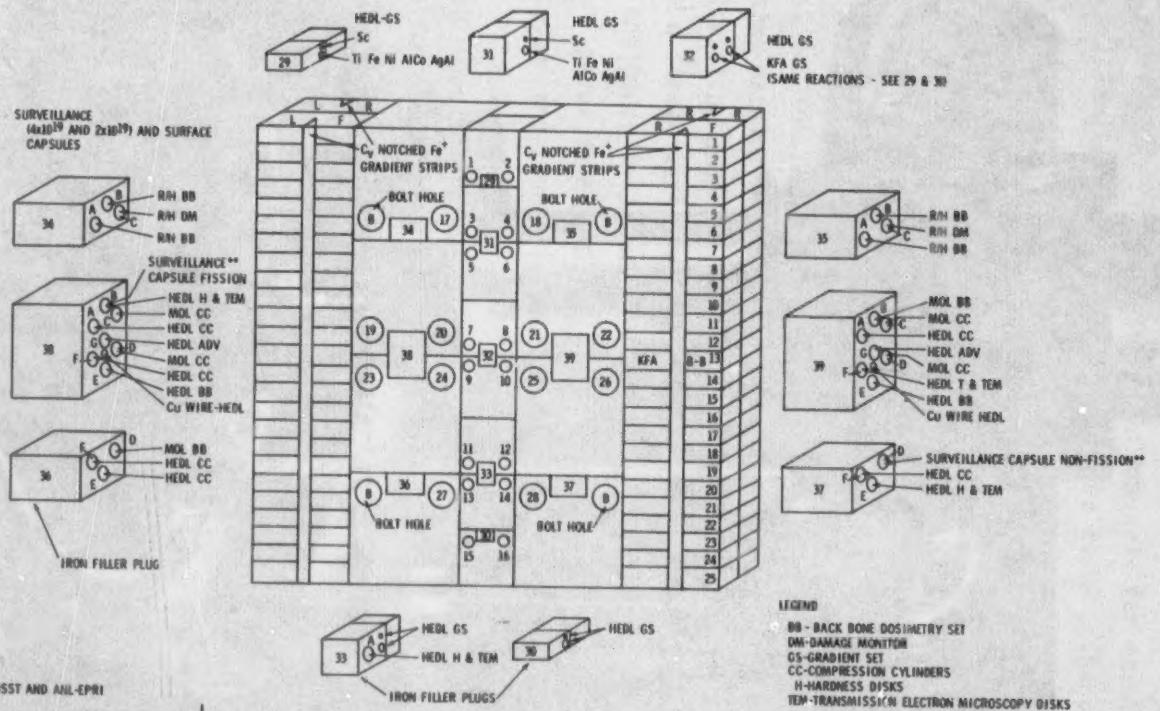


FIGURE 20. "MTS" Hydraulic Compression Testing Machine. Neg 8008646-3



[POSITION 1] SURVEILLANCE (~ 2.0 x 10^{19})*, [POSITION 1] SURVEILLANCE (~ 3.6 x 10^{19})*, AND [POSITION 2] SURFACE (~ 3.6 x 10^{19})* CAPSULES

HEDL-GS
Sc
Ti Fe Ni AlCo AgAl
29
31
HEDL GS
Sc
Ti Fe Ni AlCo AgAl
32
HEDL GS
KFA GS (SAME REACTIONS - SEE 29 & 30)
SURVEILLANCE (4x10^19 AND 2x10^19) AND SURFACE CAPSULES
34
A
B
C
R/H BB
R/H DM
R/H BB
SURVEILLANCE** CAPSULE FISSION
38
HEDL H & TEM
MOL CC
HEDL CC
HEDL ADV
MOL CC
HEDL CC
HEDL BB
Cu WIRE-HEDL
36
D
E
F
MOL BB
HEDL CC
HEDL CC
IRON FILLER PLUG
L
F
R
C_v NOTCHED Fe+ GRADIENT STRIPS
C_v NOTCHED Fe+ GRADIENT STRIPS
BOLT HOLE
BOLT HOLE
BOLT HOLE
BOLT HOLE
B
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22
23
24
25
26
27
28
KFA
B-B
35
R/H BB
R/H DM
R/H BB
39
MOL BB
MOL CC
HEDL CC
HEDL ADV
MOL CC
HEDL T & TEM
HEDL BB
Cu WIRE HEDL
37
SURVEILLANCE CAPSULE NON-FISSION**
HEDL CC
HEDL H & TEM
33
HEDL GS
HEDL H & TEM
30
HEDL GS
IRON FILLER PLUGS

LEGEND

BB - BACK BONE DOSIMETRY SET
DM-DAMAGE MONITOR
GS-GRADIENT SET
CC-COMPRESSION CYLINDERS
H-HARDNESS DISKS
TEM-TRANSMISSION ELECTRON MICROSCOPY DISKS

°SHOULD INCLUDE ORNL-HSST AND ANL-EPRI

PT#	HOLE	LAB	CONTENTS
17	RRA-	HARWELL	CC & TEM
18	"	"	"
19	AVAILABLE FOR UP TO THREE GE DOSIMETRY SETS		
20	" " " " " CE	"	"
21	" " " " " WEST.	"	"
22	" " " " " BW	"	"
23	" " " " " OTHERS°	"	"
24	$4x10^{19}$ SURV. AND SURFACE CAPS.	HEDL	EXPLORATORY SSTR-DM
25	"	"	"
26	AVAILABLE FOR UP TO THREE OTHERS DOSIMETRY SETS		
27		HEDL	H & TEM
28		"	"

PT#	HOLE	LAB	CONTENTS
29		HEDL	GS
30		"	"
31		"	"
32		HEDL & KFA	"
33		HEDL	(GS+(H&TEM IN A)
34	A	RRA/HAR (R/H)	BB
	B	"	BB
	C	"	DM
35	A	"	BB
	B	"	BB
	C	"	DM
36	D	MOL	BB

PT#	HOLE	LAB	CONTENTS
36	E	HEDL	533-CC
	F	HEDL	533-CC
37	D	HEDL	NON-FISSION
	E	HEDL	H&TEM
	F	HEDL	533-CC
38	A	HEDL	302-CC
	B	HEDL	FISSION
	C	MOL	CC
	D	MOL	CC
	E	HEDL	BB
	F	HEDL	302-CC
	G	HEDL	ADV DOS

PT#	HOLE	LAB	CONTENTS
39	A	HEDL	302-CC
	B	MOL	BB
	C	MOL	CC
	D	MOL	CC
	E	HEDL	BB
	F	HEDL	H&TEM
	G	HEDL	ADV DOS
CVRF-13		KFA	BB
CVLF Fe		NOTCHED △	
CVLR Fe		" "	
CVRF "		" "	
CVRR "		" "	

*FLUENCE (n/cm²)

**ONLY IN POSITION 1 SURVEILLANCE (~2.0 x 10^{19}) CAPSULE.

+FABRICATED BY ORNL ~0.020" DIA. FE WILL BE USED INSTEAD OF NOTCHED GRADIENT STRIPS.

HEDL 7906-170.1

FIGURE 21. Metallurgical Specimen Configuration.

TABLE 7

DESCRIPTION OF COMPRESSION CYLINDER SPECIMENS

HEDL Material Designation	Source and Source Code	Dimensions	Irradiation Locations	Material Description
27 N	NRL 27N	.157" diam. X .392" long cylinder	Located in Charpy carriers Fig (22) in LR1, LR2, and RR1 positions in SSC-1, SSC-2 1/4T, 1/2T and PVF capsules. Also in void box.	A533B plate material, cross rolled, .13% Cu, 1/4T position.
29 N	NRL 29N			A533B plate, cross rolled, .03 Cu, 1/4T position.
BG 1	EPRI 2bE			A508-2 Forging. 1/4T depth. EPRI received from B&W.
BG 2	EPRI EP24			Hi Cu-Hi Shelf. Linde 0091 weld on A533B-1 base. EPRI/West/NRL Source.
BG 3	EPRI 1bA			Originally from NRL. Surface Of HSST-02 plate. A533B.
BG 4	EPRI NP 933-1-150 Heat Code 1mQ			EPRI archive. E8018-03 electrode 1 mQMMA weld on an SA533B-1 base. Weld by Combustion Engineering. CE gave plate and weld to EPRI.
BG 5	EPRI 1bA			Originally from NRL. Broken halves of 1TCT specimens differ from BG 3 in that it is 1/4T location. HSST-02 A533B plate.
BG 6	EPRI 4bA			Originally from NRL. Broken halves of 1TCT specimens A302B ASTM heat, samples from 1/4T position.
BG 7	EPRI 7bB			A533-2 material went to EPRI from General Atomic, Manufactured by Lukens. Broken halves of 1TCT specimens.
3 PT	NRL-3PT	.250" diam. X .410" long cylinder	Located in block holes (Fig 21) 38A, 38F, 39A, 36F, 36E, 37F of SSC-1, SSC-2, 1/4T, 1/2T and and PVF capsules	HSST-03 plate of A533B. 1/4T location cut directly from plate.
F 23	NRL-F23			ASTM Reference heat 1/4T location cut directly from plate.

NOTE: Accumulation of above materials was done principally by Fracture Control Corp. (R. Wullaert)

TABLE 8

COMPRESSION CYLINDER CHEMISTRY (wt %)

HEDL Designation	C	Mn	P	S	Si	Ni	Cr	Mo	V	Cu	NDT	RTNDT	Upper Shelf
BG-1	.23	.73	.007	.009	.31	.65	.41	.59	.054	.04	-7°C	-7°C	182.J
BG-2*	.11	1.57	.007	.011	~.23	.64	.02	.48	.005	.36	---	---	---
BG-3	.23	1.55	.009	.014	.2	.67	.04	.53	.003	.16	-30°C	-15°C	144.J
BG-4	.1	1.11	.007	.01	.4	1.06	.01	.34	.006	.02	-57°C	-57°C	212.J
BG-5	.23	1.55	.009	.014	.2	.67	.04	.53	.003	.16	---	---	---
BG-6	.23	1.47	.013	.024	.26	.17	.05	.52	.004	.2	-18°C	---	61.J
BG-7	.21	1.25	.007	.007	.170	.210	.250	.08	.005	.12	---	---	138.J
27 N	.18	1.29	.008	.008	.20	.56	.06	.51	---	.13	---	---	---
29 N	.18	1.27	.008	.007	.20	.55	.06	.52	---	.03	---	---	---
3 PT	.20	1.26	.011	.018	.25	.56	.10	.45	---	.10	---	---	---
F 23	.23	1.47	.013	.024	.26	.17	.05	.52	.004	.20	---	---	---

* Preliminary analysis

TABLE 9

CHARPY, TENSILE, AND CT SPECIMENS FOR EACH OF FIVE CAPSULES FOR PSF*

Metallurgical Material	Regular** Charpy V	Charpy** Sized Tensile Specimen	0.5TCT	1.0TCT
EPRI Weld Metal	12	2	0	0
NRC/NRL A302B Ref. Plate	18	5	10	5
NRC/NRL A553-03 Ref. Plate	10	2	6	3
KFA/22Ni Mo Cr37, KFA, Jülich, Germany	14	2	0	0
CEN-SCK/COCKERILL SA 508 CLASS-3 FORGING, CEN/SCK, Mol, Belgium	14	2	0	0
High Sensitivity SA Weld (.23 Cu) on an A533 Base, Rolls-Royce and Associates Limited, United Kingdom	12	2	0	0

*Specimens and materials for the void box position are not identified in this table.

**The accounting of Charpy positions is that 95 Charpy or Charpy equivalent spaces were allotted to participants and five such spaces were available for space compatible specimens, for utilization by assemblers for dosimeters, or for other uses.

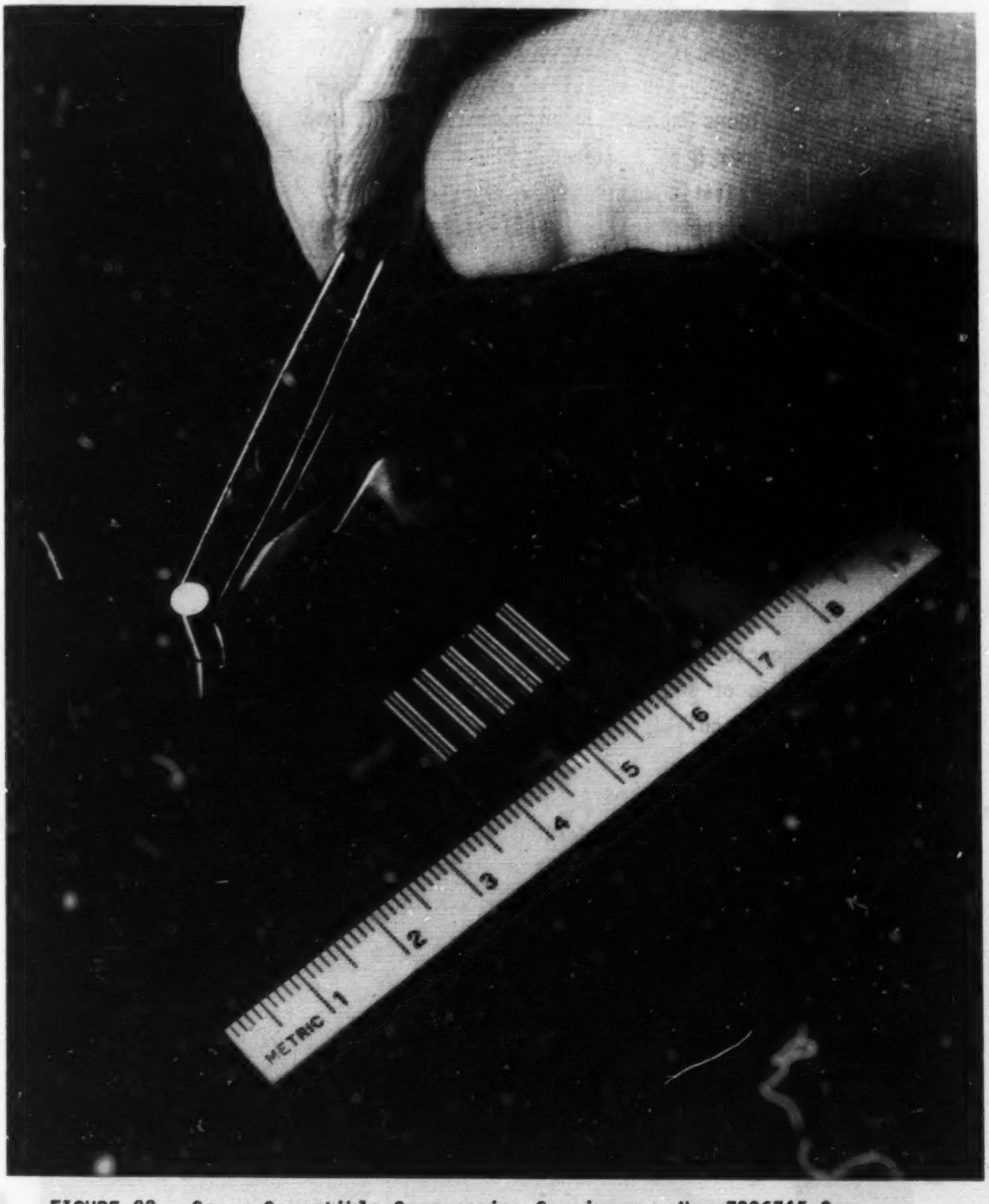
TABLE 10

A. SPACE COMPATIBLE SPECIMEN MATERIALS FOR PSF (ENGINEERING MATERIALS)

Material	Heat Code	Source	Description
A302B Ref. Plate	F23	NRC/NRL	Table 8 Material
A533-03 Ref. Plate	3PT	NRC/NRL	Table 8 Material
A508	-	CEN/SCK, Mol, Belguim	Forging
A533B-1	1bA	EPRI Archive (HSST-02)	1/4T, 4-1T Compact Halves
A533B-1	1bA	EPRI Archive (HSST-02)	Surface, 6 1/2x10x2 1/4 Piece
A302B	4bA	EPRI Archive (ASTM-CM)	1/4T, 4-1T Compact Halves
SA Weld-Linde 0091 (A533B-1 base)	EP24	EPRI/West/NRL	Hi Cu, Hi Shelf
MMA Weld-E8018-C3 (A533B-1 base)	1mQ	EPRI Archive (CE)	11 x 13 x 6 Piece
A508-2	2bE	EPRI/Archive (B&W)	1/4T, 5 x 9 x 1 Piece
A537-2	7bB	EPRI/Archive (GA)	1/4T, 4-1T Compact Halves

B. SPACE COMPATIBLE SPECIMEN MATERIALS FOR PSF (MODEL MATERIALS)

Material	Heat Code	Source	Description
A533B-1 (0.03Cu)	N29	NRL/Hawthorne	1/4T, 3 pieces 1/2 x 1/2 x 1/4 in
A533B-1 (0.13Cu)	N27	NRL/Hawthorne	1/4T, 3 pieces 1/2 x 1/2 x 1/4 in
Pure Iron	Fe-6	NRL/Smidt	30 TEM Discs
Fe - 0.3at%V	Fe-2	NRL/Smidt	30 TEM Discs
Fe - 0.3at%Cu	Fe-4	NRL/Smidt	30 TEM Discs
Fe - .34wt%Cu	Alloy 1	West./Spitznagel	Misc. Pieces of Strip & Wire
Fe - 0.1wt%N	Alloy 2	West./Spitznagel	Misc. Pieces of Strip & Wire
Fe - .34wt%Cu - 0.1wt%N	Alloy 3	West./Spitznagel	Misc. Pieces of Strip & Wire


METRIC
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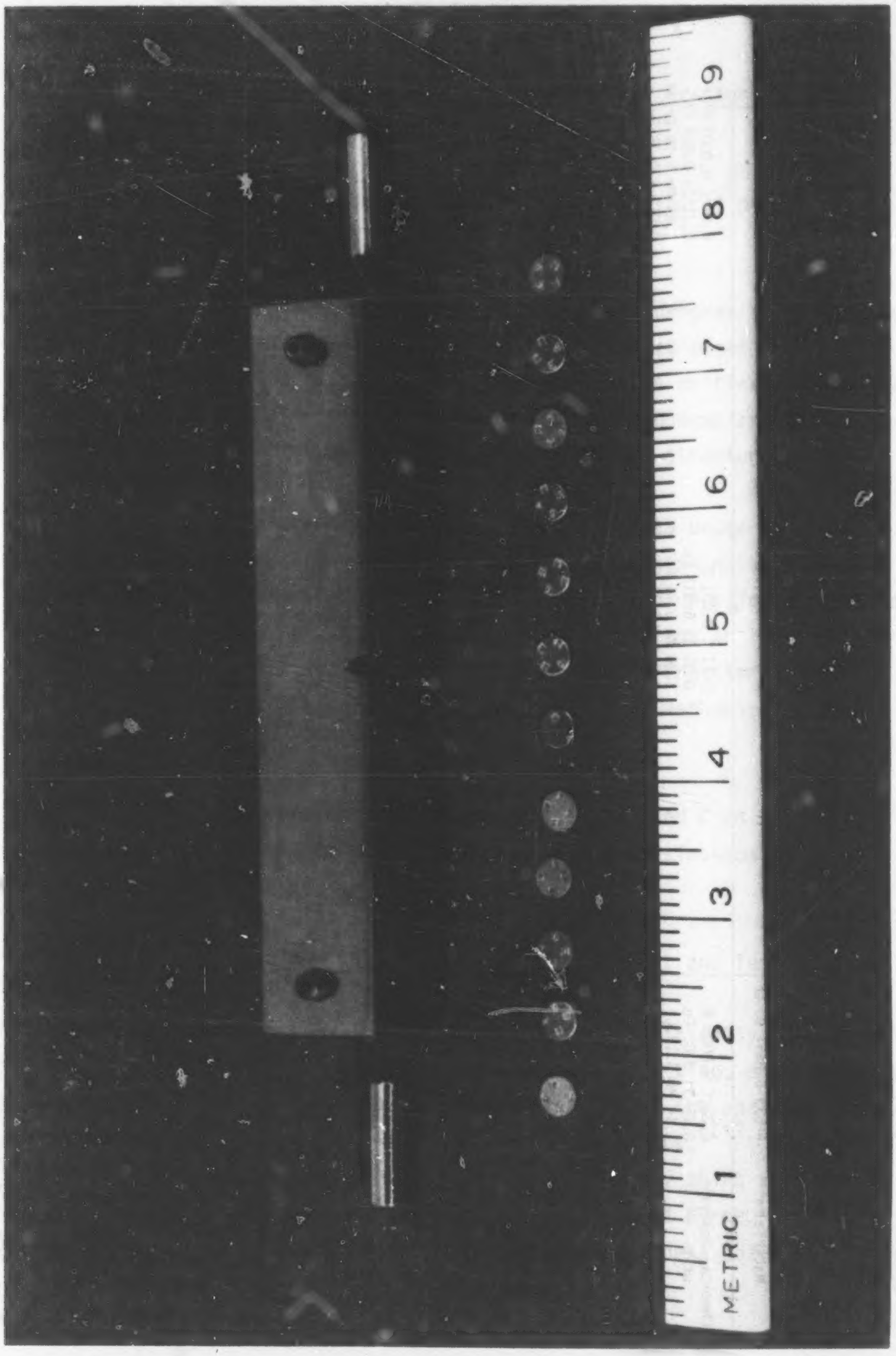
8

FIGURE 22. Space Compatible Compression Specimens. Neg 7906745-2cn

California at Santa Barbara, EPRI, Fracture Control Corporation and HEDL personnel using "engineering" and "model" materials obtained from a number of different sources identified in Table 10.

Two types of holders in each irradiation capsule contain small disks 0.003" to 0.004" thick and 3 mm diameter. These holders are 1) Charpy carriers (one in each irradiation capsule; at the surveillance, surface, 1/4T, and 1/2T locations as shown in Figure 23 and 2) small open cylindrical holders which fit in holes in the transverse drilled blocks (e.g., block 38F of Figure 21). These specimens can be used in TEM examinations and in hardness studies. Some preliminary work has been done by L. Thomas at HEDL to prepare for the TEM examinations.(26) The hardness examinations are expected to be conducted using commercial hardness test machines together with instrumented hardness test procedures and microhardness techniques under development by G. Odette and G. Lucas at the University of California at Santa Barbara.

The preliminary TEM studies mentioned above have been conducted as part of a joint KFA-HEDL study of samples of three alloy heats irradiated in West German test reactors. This study involves TEM examination and coordinated hardness studies of samples subjected to a series of isochronal anneals. The hardness data collected for this study are already quite extensive(39) but the TEM studies are in a preliminary stage. To date, the HEDL TEM part of the program has consisted of studies of methods of electropolishing the samples, as well as TEM examinations of unirradiated control specimens and irradiated specimens prior to initial isochronal anneals. A list of the joint HEDL-KFA specimens available for examination appears in Table 11.


METRIC
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6
7
8
9

FIGURE 23. Space Compatible Hardness and TEM Specimens. Neg 7906745-6cn

TABLE 11

PROPOSAL FOR TEM ANNEALING HEAT TREATMENT TEMPERATURES FOR ISOCHROMAL ANNEALS

Material	A 533 B HSST 03 LF					A 533 B Modified					Weld
Chemical Analysis wt %	0.22 C 0.0005 B 0.25 Si 0.01 Co 1.44 Mn 0.12 Cr 0.012 P 0.12 Cu 0.015 S 0.51 Mo 0.023 N 0.63 Ni 0.023 Al 0.01 V					0.11 C 0.001 B 0.35 Si 0.013 Co 1.36 Mn 0.11 Cr 0.012 P 0.11 Cr 0.008 S 0.37 Mo 0.016 N 1.5 Ni 0.014 Al 0.015 V					0.06 C 0.0003 B 0.21 Si 0.023 Co 1.85 Mn 0.08 Cr 0.012 P 0.29 Cu 0.003 S 0.56 Mo 0.011 N 1.3 Ni 0.015 Al 0.02 V
Capsule No.	KFA 1	KFA 2	KFA 3	KFA 4	KFA 5	KFA 6	KFA 7	KFA 8	KFA 9	KFA 10	KFA 11
Neutron Fluence n/cm^2 E 1 MeV	1.5×10^{18}	1.5×10^{19}	1.5×10^{19}	3×10^{19}	7×10^{19}	1.5×10^{18}	1.5×10^{19}	1.5×10^{19}	3×10^{19}	7×10^{19}	5×10^{19}
Neutron Flux FRJ-1 n/cm^2 E 1 MeV	3×10^{13}	3×10^{13}	6×10^{11}	3×10^{13}	3×10^{13}	3×10^{13}	3×10^{13}	6×10^{11}	3×10^{13}	3×10^{13}	3×10^{13}
Transition Temp. TT 30 Ft-Lb=51 J/cm^2	60°C	145°C	150°C	145°C	187°C	24°C	110°C	105°C	114°C	168°C	--
Increase of Transition Temp. ΔTT	35°C	120°C	125°C	120°C	162°C	36°C	122°C	117°C	126°C	180°C	--
Vickers Hardness HV 10 N/mm^2	2200	2800	2700	2820	2980	2250	2800	2840	2930	3070	3440
Increase of Vickers Hardness	10%	40%	35%	41%	50%	23%	53%	55%	60%	68%	55%
Irradiation Temp.	150°C	150°C	150°C	150°C	150°C	150°C	150°C	150°C	150°C	150°C	150°C
Proposal for TEM Annealing Heat Treatment Temperature for Isochromal Anneals 1) 2) 3) 4) 5) 6)	A.R. 330°C 400	A.R. 260 310 400 490	A.R. 280 330 400 490	A.R. 260 310 400 490	A.R. 270 310 400 490	A.R. 310 400 490	A.R. 270 330 400 490	A.R. 290 330 400 490 600	A.R. 270 330 400 490 600	A.R. 270 330 400 490 600	A.R. 290 330 400 500

NOTE: A.R. = As Received, i.e., not yet annealed.

VI. Analysis and Interpretation of Research and Power Reactor Test and Surveillance Results

[Variables Studied: All those initially listed in the first paragraph of Section II.]

A primary objective of this multilaboratory supported program is to help in the development of statistically valid radiation embrittlement data bases (Reg. Guide 1.99, MPC, EPRI, and others) for use in the critical evaluation of the procedures used for predicting the fracture toughness and embrittlement of irradiated reactor pressure vessel and support structure steels.

Analysis of the existing and new additions to these data bases (from test and power reactors) has revealed that the variance of test data does not arise entirely from material variability. A substantial portion stems from lack of consistency in the application and/or shortcomings in test methods and control of important variables associated with the "reactor systems analysis," "dosimetry," "metallurgical," and "fracture mechanics" disciplines.

Summary information is presented in Sections VI-A, B, and C on the results of recent studies associated with metallurgical and physics-dosimetry data development and testing.

A. Surveillance Capsule Metallurgical Data Development and Testing

As part of the LWR Dosimetry Improvement Program, statistically based data correlation studies have been made by G. Guthrie, F. Berting, and R. Simons of existing metallurgical data banks in anticipation of the analysis of new embrittlement data from the ORR-PSF and other experiments.(1,2,6-9,25-27,44-47) Some of the available data collections are 1) the Charpy shift vs exposure data for welds and plates, as collected by Beimiller and Byrne,(42) and Varsik and Byrne,(43), 2) the Charpy shift vs exposure data collected by the MPC study(38), and 3) the data available in various reactor surveillance reports. The immediate purpose of such analyses is the application of the results to the preparation of the ASTM Standard Practice, IE, on "Damage Correlation" and the ASTM Standard Guide, IIF, on "Correlation of NDTT with Fluence," Figures 1 and 2.

Nineteen surveillance capsule reports for Westinghouse reactors have been examined for inconsistencies in reported fluence values. Details of this work are discussed in Section VI-B-4 and summary results are presented in Table 16. Twelve of the 19 capsules analyzed contained Charpy specimens from the so-called ASTM A302B "reference heat." As a first approximation, it is possible to assume that the chemistry of these specimens is roughly identical. Therefore it is possible to use these data to check for the applicability of various forms of equations relating ΔNDTT and flux or fluence, while avoiding the complexities usually imposed by chemical variations.

Several formulas were used to relate ΔNDTT to flux or fluence. Of the expressions examined, it was found that a formula of the type

$$\Delta \mathrm{NDTT} = A(\phi t)^{N} \tag{1}$$

gave the best fit for the set of 12 Charpy shift data points. The 1σ standard deviation of the fit was 11.45°F with $A = 261$ and $N = 0.238$ when ΔNDTT is expressed in $^{\circ}$F and ϕt is given in DPA.

Another satisfactory expression was

$$\Delta \mathrm{NDTT} = A \cdot \left[\tau \cdot \phi \cdot (1 - \exp(-\frac{t}{\tau})) \right]^{N} \tag{2}$$

where ΔNDTT is given in $^{\circ}$F, and the derived values were $A = 282$, $\tau = 2.73 \times 10^{8}$ sec, and $N = 0.248$, and t is given in seconds while ϕ is expressed in DPA/sec. For this relation, $\sigma = 12.11^{\circ}$F. It might be noted that τ is 8.66 years which would indicate the presence of a saturation process which would only become apparent with data from long term irradiations.

Simple theories relating resistance of dislocation flow to obstacle density would indicate that the exponent on the bracketed term of Eq (2) might well be set at the value 0.5. This would be in harmony with the point of view that the bracketed term is somehow associated with the number of remaining defects at any given time and the exponent relates the defects to the flow resistance. However, this also assumes a simple proportionality between flow resistance and ΔNDTT. The use of such a formula, with N set equal to 0.5 in Eq (2), however, did not give improved results. The value derived for sigma was 23.09°F.

Other formulas tested included

$$\Delta NDTT = A \cdot [1 - \exp(-B \cdot \phi t)]^N \tag{3}$$

which gave a value of sigma of the order of 12°F, and

$$\Delta NDTT = A \cdot \left[1 - \exp\left(-\frac{t}{\tau_1}\right)\right] + B \cdot \left[\phi \cdot \tau_2 \cdot \left(1 - \exp\left(-\frac{t}{\tau_2}\right)\right)\right]^{0.5}. \tag{4}$$

This latter formula gave a sigma of approximately 18°F.

The use of a simple straight line relation,

$$\Delta NDTT = A + B \cdot (\phi t) \tag{5}$$

gave a value of sigma of 12.22°F.

The statistical similarity of results with and without saturation effects, Eq (1) and Eq (2), indicates no apparent advantage for one equation over the other. However, when the two formulas are extrapolated to a 30-year end-of-life type irradiation, Eq (1) shows Charpy shifts approximately one third larger than those given by Eq (2).

The advantages to be found by removing inconsistencies from existing data banks are shown by taking the 12 data sets for the 12 capsules, both in the corrected and uncorrected forms, and applying a mathematical relationship of the type given by Eq (1). This results in a decrease in sigma of approximately 2.0°F in going from the uncorrected to the corrected values. The lack of any startling improvement is due to the fact that the error in the determination of the Charpy 30 ft-lb transition temperature is approximately 10°F, which places a lower limit on the value of sigma. Also, the data are all taken from one type of PWR plant where reasonably uniform calculational procedures, cross-section values, etc. have been used in the analysis of the data. Therefore the errors tend to be consistent and incorrect values show a bias rather than a scatter.

An analysis of the MPC data base was performed to detect features in the data which would be indicative of the presence of fluence reporting errors.(27) Such an analysis is possible because fluence errors tend to follow a log-normal distribution rather than a normal one. The results of the analysis showed that a reduction of 7°F in sigma can be expected if fluence errors are removed, but this does not take into account the additional benefits derived from the removal of the bias.

B. Surveillance Capsule Dosimetry Data Development and Testing

1. Program Status

In establishing improved metallurgical data bases, careful consideration must be given to the variables associated with the use and accuracy of neutron exposure units and values. The total fluence above 1.0 MeV is currently used as the critical spectral parameter for the prediction of LWR reactor material property change. This is a widely accepted practice, even though the mechanisms which effect the radiation-induced property change cannot be described in such simple terms. The use of the DPA as a more realistic exposure unit for the measurement of materials property change has been recommended and an ASTM Standard E693-79 is now available for its routine application,(49) Figure 1.

The effects of using different exposure units, of neutron spectrum changes from surveillance through ex-vessel positions, of photofission and surveillance capsule flux perturbations, and the impact of the selection of RM, SSTR, HAFM, and Damage Monitor (DM) dosimetry sets on the accuracy of derived exposure values (with and without "Benchmark Field Referencing"), are discussed in references 1, 2, 4, 5, 10, 11, 23, 28, 29, and 48. In this section of this report emphasis is placed on a discussion of the results of validation and calibration studies associated with current state-of-the-art LWR surveillance capsule dosimetry measurements and derived exposure values.

Table 12, taken from reference 2, provides the results of a previous study of individual and multiple foil reported exposure values for several PWR and one BWR power plant. The results were obtained from several surveillance capsule reports and showed the existence of serious discrepancies between fission and nonfission radiometric monitor results.(2) In a number of cases the fission foil fluence values were up to a factor of 1.5 to 2.7 higher than the nonfission foil results. In addition to data reduction and radiometric monitor measurement problems, these discrepancies are thought to be associated with surveillance capsule flux perturbation and photofission effects.(1,2,10,23,28,29,48)

2. ORR-PSF Dosimetry/Perturbation Experiment

For a Westinghouse surveillance capsule geometry for a 2-loop PWR power plant (at the center of an approximately 1" x 1" x 48" bar of steel located behind the thermal shield), S. Anderson has calculated perturbed $^{54}Fe(n,p)^{54}Mn$ reaction rates that are approximately 12 to 15% less and $^{237}Np(n,f)F.P.$ fission rates that are approximately 40 to 60% higher than the unperturbed values for 13°, 23°, and 33° azimuthal positions. Consequently, without correcting for these perturbation effects, the $^{54}Fe(n,p)^{54}Mn$ and $^{237}Np(n,f)F.P.$ reported values of flux greater than 1.0 MeV could appear to be discrepant by approximately 73%, 55%, and 64% for the 13°, 23°, and 33° azimuthal positions, respectively.(1,48)

TABLE 12

REPORTED SURVEILLANCE CAPSULES SINGLE FOIL FLUX/FLUENCE VALUES [ϕ_i >1 MeV] - RELATIVE TO ^{54}Fe(n,p)

Reactor Name (Vendor-Type, Country, Operation Date)	Service Laboratory Report	SINGLE FOIL FLUX/FLUENCE >1MeV			
		$\phi_{^{58}Ni}(n,p)$	$\phi_{^{63}Cu}(n,\alpha)$	$\phi_{^{238}U}(n,f)$(e)	$\phi_{^{237}Np}(n,f)$(e)
Point Beach #1	BMI (1973)	1.09	1.63	1.61	2.17
(West. PWR, USA, 12/70)		[Reported Surveillance Value: 1.0(Fe)](d)			
Same (Angle A)	WEST. (1979) (a)	0.80	1.51(b)	1.03	1.25
Same (Angle A + 180°)	WEST. (1979) (a)	1.01	1.14(b)	1.13	1.01
		[Reported Surveillance Value: 1.0 (Fe)](d)			
Average Values For Seven WEST. Power Plants	WEST. (1979) (a)	0.97	1.25(b)	1.08	1.15
Humboldt Bay 3	GE (1967)	0.88	0.80	-	-
(GE BWR, USA, 8/63)		[Reported Surveillance Value: 1.0 (Fe)](d)			
San Onofre #1	SWRI(1971)	1.00	1.27	1.10	1.42
(West. PWR, USA, 1/68)					(BNW Spectrum 1)
(West. PWR, USA, 1/68)	SWRI(1971)	1.05	0.88	1.29	1.45 (BNW Spectrum 2)
		[Reported Surveillance Value: 0.85 (SAND II, multiple foils)](d)			
OCONEE #1	B&W (1975)	1.18	-	2.50	2.70
(B&W PWR, USA, 7/73)		[Reported Surviellance Value: $\frac{1.0 + 2.5}{2}$ = 1.76 (Fe+U)](d)			
DOEL I	CEN/SCK (1979)	1.09	1.51(b)	-	2.41
Belgium-West. PWR Belgium 1/75)	CEN/SCK (1979)	1.09	1.06(c)	-	2.41
		[Reported Surveillance Value: ∿ 1.09 (Ni)] (d)			

TABLE 12 FOOTNOTES

(a) Surveillance capsule flux perturbation corrections were calculated by Westinghouse to provide necessary correlations between the U, Np, Cu, Ni, and Fe results. None of the other results shown in the table have been corrected for perturbation effects.

(b) ENDF/B-IV $\sigma(E)$ for $^{63}Cu(n,\alpha)$

(c) Mann-Schenter $\sigma(E)$ for $^{63}Cu(n,\alpha)$

(d) These reported surveillance capsule measured fluence values are used for correlating the surveillance capsule metallurgical data with other test and power reactor data. They are also used for making localized predictions of expected pressure vessel lifetime neutron exposures and/or can be used to simply confirm the correctness of one, two, and three dimensional reactor physics computations.

(e) Based on ^{137}Cs analysis.

NOTE: The Ni, Fe, and Cu provide experimental fluence data for time periods up to about 1 year, 5 years, and 25 years with a knowledge of the surveillance capsule flux level time history, this information is not needed for a reliable interpretation of fission foil $^{137}Cs(t_{1/2} \sim 31$ years) results.

To provide experimental verification of these and other analytical predictions, a dosimetry perturbation experiment was irradiated in the PSF high-power LWR pressure vessel mockup facility. Two 1" x 1" x 12" long, carbon-steel mockup surveillance capsules, supplied by Westinghouse, were positioned in the PSF for a nine-day irradiation. Both capsules were exposed simultaneously to a maximum fluence of about 5 x 10^{18}n/cm^2 (E > 1 MeV). One capsule was located at the outer surface of the thermal shield, six inches to the right of center, and the other at the pressure vessel surface six inches to the left of center. The PSF is a 4/12 configuration. Radiometric monitors (RM), solid state track recorder (SSTR), and helium accumulation fluence monitors (HAFM) were positioned within the capsules by HEDL and CEN/SCK. Six replicate dosimeter sets were irradiated in nearly zero gradient configurations so that essentially identical dosimeters were supplied for independent analysis by four vendors (W, GE, CE, and B&W) and two services laboratories (SWRI and BMI). Analysis and "Benchmark Field Referencing" of these and other RM, SSTR, and HAFM monitor results is being accomplished by CEN/SCK, EG&G-ORTEC, HEDL, NBS, ORNL and RI. The six sets of dosimeters were recovered and shipped to HEDL for initial radiometric analyses. Subsequently, one complete set of dosimeters was sent to each of the six laboratories for counting; final counting and data analysis is in progress.

A preliminary analysis of laboratory-to-laboratory biases for the ORR-PSF Dosimetry/Perturbation Experiment for the radiometric monitors has been completed by L. Kellogg and A. Davis. The Table 13 evaluation shows the present laboratory-to-laboratory comparative status but also shows the improvements in the data comparisons as a result of interim evaluations and discussions with participants. All laboratories remain anonymous for these intercomparisons and are identified only as laboratories A, B, C, D, E, and F. With reference to Table 13, and for each reaction listed, the following information is provided:

TABLE 13

RANGE EVALUATION (MAXIMA/MINIMA) RADIOMETRIC RESULTS

	^{58}Ni(n,p)				^{46}Ti(n,p)				^{63}Cu(n,α)				^{54}Fe(n,p)			
	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs
HWF - 1	1.14 C/B	1.10 D/B	1.06 D/C	6	1.39 C/B	1.16 C/B	1.04 C/E	6	1.11 F/B	------	1.11 F/E	6	1.38 C/B	1.11 C/B	1.04 C/E	6
- 3	1.28 C/B	1.06 A/C	1.05 A/C	6	1.48 C/B	1.21 C/B	1.07 C/E	6	1.05 C/E	1.06 C/F	------	6	1.27 C/B	1.10 C/B	1.04 C/F	6
- 2	1.25 C/B	1.13 F/B	1.06 F/E	6	1.42 C/B	1.19 A/B	1.08 A/E	6	1.06 C/E	1.09 C/E	------	6	1.46 C/B	1.12 C/B	1.07 C/E	6
- 4	1.28 C/B	1.14 F/B	1.06 F/E	6	1.41 C/B	1.17 C/B	1.10 C/E	6	1.08 C/E	1.07 C/E	------	6	1.43 C/B	1.15 C/B	1.09 C/F	6

	^{58}Fe(n,γ)			
	Ratio 1	Ratio 2	Ratio 3	Labs
HWF - 1	1.15 C/A	1.03 E/A	1.03 E/A	3*
- 3	1.02 C/E	1.02 A/E	1.02 A/E	3
- 2	1.23 C/A	1.04 E/A	1.04 E/A	3
- 4	1.12 C/E	1.01 A/E	1.01 A/E	3

	^{59}Co/Al(n,γ)			
	Ratio 1	Ratio 2	Ratio 3	Labs
HF - 3	1.15 C/B	1.09 C/B	1.06 C/F	6
- 5	1.15 C/B	1.11 C/B	1.04 C/A	6
- 4	1.06 F/B	1.10 C/B	1.05 C/D	6
- 6	1.23 C/B	1.17 C/B	1.07 C/E	6

	^{235}U(n,f)^{140}Ba				^{235}U(n,f)^{103}Ru				^{235}U(n,f)^{95}Zr				^{235}U(n,f)^{137}Cs			
	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs
HF - 3	1.06 A/E	------	------	3	1.19 C/D	1.22 A/C	1.10 A/D	4	1.09 D/C	1.22 D/C	1.08 D/E	5	------	1.11 E/C	------	3
- 5	1.19 A/D	------	------	3	1.73 C/D	1.19 A/D	1.07 A/E	4	1.51 C/B	1.29 C/B	1.12 C/E	5	------	1.31 C/D	------	3
- 4	1.05 A/D	------	------	3	1.30 C/D	1.11 A/C	------	4	1.13 D/C	1.14 D/C	1.07 A/C	5	------	1.06 E/C	------	3
- 6	1.10 A/D	------	------	3	1.43 C/D	1.14 A/C	------	4	1.15 C/E	1.11 A/C	1.07 A/B	5	------	1.07 E/D	------	3

	^{237}Np(n,f)^{140}Ba				^{237}Np(n,f)^{103}Ru				^{237}Np(n,f)^{95}Zr				^{237}Np(n,f)^{137}Cs			
	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs
HF - 1	1.10 D/F	------	------	4	1.37 C/F	1.11 A/F	------	5	1.16 A/E	1.21 A/C	1.16 A/E	6	------	1.09 D/C	------	4
- 2	1.13 A/E	------	------	4	1.27 C/F	1.12 A/F	------	5	1.14 D/F	------	------	6	------	1.14 F/D	------	4

	^{238}U(n,f)^{140}Ba				^{238}U(n,f)^{103}Ru				^{238}U(n,f)^{95}Zr				^{238}U(n,f)^{137}Cs			
	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs	Ratio 1	Ratio 2	Ratio 3	Labs
HF - 1	1.09 A/E	------	1.04 B/E	4	1.44 C/F	1.16 A/F	------	5	1.16 C/F	1.09 B/F	------	6	------	1.08 E/C	------	4
- 2	1.10 A/E	------	------	4	1.27 C/F	1.13 A/F	------	5	1.13 D/F	------	------	6	------	1.08 B/C	------	4

*Results for the ^{58}Fe(n,γ) reaction were not reported by one laboratory after preliminary recalibration of their counting systems.

a. Ratio 1 - The maxima/minima range (within set) of the reported results from the participating laboratories--after initial review of the data to eliminate obvious errors resulting from the handling of a large number of samples in a relatively short time period.

b. Ratio 2 - The maxima/minima range after preliminary recalibration of one laboratory.

c. Ratio 3 - Nonfission monitors--The maxima/minima range after eliminating the results from one laboratory that appears to be consistently biased low on the majority of the reported activitation products. (Investigation into this possible bias is being made by the participant). For fission monitors--the maxima/minima range after elimination of an apparent high or low result to indicate the general overall range.

d. Labs - The number of laboratories (A, B, C, D, E, and F) reporting results utilized in the determination of Ratios 1 and 2.

The maxima/minima ratios of Table 13 show the laboratory-to-laboratory biases that currently could be introduced into surveillance capsule reported values of fluence (E > 1 MeV) as a result of the measurement and data reduction of individual flux/fluence monitors by these six laboratories. For example, if laboratories C and B were both to have reported flux/fluence values based on the ^{54}Fe(n,p)^{54}Mn reaction, the laboratory C values would have been up to a factor of 1.46 higher than the laboratory B results. With "Benchmark Field Referencing," however, such large factors as 1.27 to 1.46 for the ^{54}Fe(n,p)^{54}Mn monitor, the Ratio 1 column, would be reduced to values of 1.15 and less, the Ratio 2 and 3 columns of Table 13.

3. Benchmark Field Referencing

The bases for benchmark field referencing are irradiations performed for the purpose of detector calibrations in a neutron field of known energy spectrum and intensity. In addition to benchmark referencing the exvessel cavity dosimetry being accomplished at the Arkansas Power and Light Company Unit #1 PWR, there are two additional benchmark efforts associated with the LWR-PV Surveillance Dosmietry Improvement Program.

One is to help provide accuracy limits on the experimental data that are the basis for comparison of calculated and measured neutron flux levels and the core power in the PCA/LWR pressure vessel mockup at ORNL. This standardization program includes special certified-fluence irradiations of dosimetry materials and detectors, as well as measurements with the NBS dual fission chambers in the pressure vessel mockup. Results of this effort are discussed in Section IV.

The other major benchmark referencing activity is attempting to establish interlaboratory consistency in measuring and evaluating fluence detectors. To accomplish this, NBS periodically issues counting standards internationally. Such standards are typical dosimetry materials made radioactive by an irradiation to a certified fluence. These are normally distributed as "blinds" and participating laboratories must deduce the fluence. The first such exercise for the LWR-PVS-Dosimetry Improvement Program is drawing to a close and some preliminary results follow.

Table 14 gives the logistics of the distribution of 26 nickel foils activated in the NBS U-235 standard fission spectrum. The radioactivity of interest is the 811-keV gamma radiation from the $^{58}Ni(n,p)^{58}Co$ reaction. A 76% laboratory response was achieved and the ratio of the reported results to the NBS values are summarized in Table 15. In general, the consistency level is fairly good, with 47% of the ratios within the range 0.980-1.020. The distribution is skewed in that 68% lie in the range 0.979-1.070. A detailed report of the intercomparison will be made after participants have been notified of results and several laboratores have been given the opportunity to evaluate a second nickel counting standard. All laboratories remain anonymous in these intercomparisons, as they do for the ORR-PSF Dosimetry Perturbation Experiment.

TABLE 14

LOGISTICS OF THE CERTIFIED FLUENCE IRRADIATIONS OF NICKEL FOILS IN THE NBS CAVITY FISSION (U-235) SPECTRUM

NICKEL FOIL IDENTIFICATION	DATES AND TIMES OF IRRADIATION START	END	TIME DURATION (SECONDS)
BV, BW, BX, BY	7/17/78 17:19 EDST	7/18/78 23:27 EDST	108480
CA, CB, CC, CD	8/1/78 17:35 EDST	8/2/78 23:10 EDST	106500
AP, AR, AS, AT, AU	8/21/78 20:51 EDST	8/22/78 17:53 EDST	75720
CF, CG, CH, CI	9/15/78 11:36 EDST	9/16/78 09:50 EDST	80040
CJ, CK, CL, CM	9/22/78 15:24 EDST	9/23/78 22:37 EDST	112380
BK, BL, BM, BN	1/13/79 14:55 EST	1/14/79 23:03 EST	115680

The foils were distributed to participants between 27 November 1978 and 3 February 1979. The percentage returned as a function of time after February 1979 is as follows:

ELAPSED TIME (months)	DATE	PERCENT RESPONSE
6	August 1979	41%
12	February 1980	69%
18	August 1980	76%

TABLE 15

COMPARISON OF RESULTS OF INTERLABORATORY CONSISTENCY IN MEASURING THE NICKEL CERTIFIED FLUENCE STANDARDS

Result Number	Ratios of Results To NBS Values
1	0.979
2	1.018
3	1.015
4	1.112
5	1.134
6	0.993
7	1.033
8	0.936
9	0.946
10	0.982
11	1.065
12	0.990
13	0.932
14	1.062
15	1.011
16	1.011
17	1.000
18	0.932
19	1.044

Overall average 1.0103 ± 0.0565 (±5.6%)

With reference to the Section VI-B-2 example (Table 13, results), if the ^{58}Ni(n,p)^{58}Co measured reaction rate had been used instead of the ^{54}Fe(n,p)^{54}Mn to obtain flux/fluence values for a surveillance capsule, the laboratory C values would have been up to a factor of 1.28 higher than the laboratory B results. Similarly, for the 19 laboratories, Table 15, the laboratory 5 values would have been a factor of 1.22 higher than the laboratory 13 and 18 results. Consequently, for the development of reliable metallurgical data bases, the range of laboratory-to-laboratory reported maxima/minima exposure values for a given irradiation experiment should be reduced and maintained at bounds that are well within a 5 to 15% (1σ) accuracy limit. In this regard, a pressure vessel benchmark facility for power reactor surveillance dosimetry validation and certification is being established at the Oak Ridge Bulk Shielding Reactor (BSR) facility. It will be identified as the BSR simulated dosimetry measurement facility (SDMF).

The information obtained from this BSR-SDMF facility will be applied to the investigation of results obtained from current surveillance capsules, so that the dosimetry methods currently applied by vendors and service laboratories can be:

1) Validated and certified;
2) Improved by development of supplementary experimental data; and
3) Evaluated in terms of actual uncertainties.

A preliminary design, shown in Figure 24 is currently being reviewed and evaluated by program participants. The facility support structure is being designed to accommodate a PVS and void box simulator, but only a thermal shield simulator is considered necessary for most applications. Initially, CEN/SCK, EG&G ORTEC, HEDL, NBS, and ORNL and a number of other program participants will cooperate in the design characterization and calibration of this facility.

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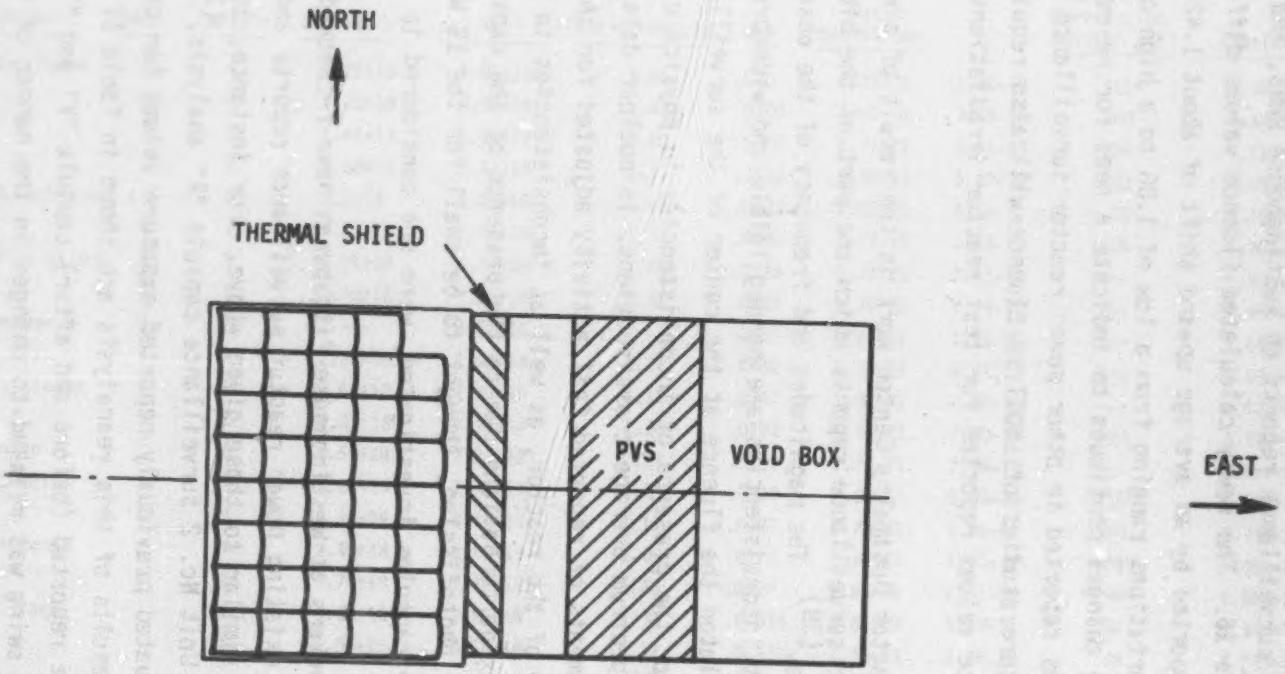
NORTH
THERMAL SHIELD
PVS
VOID BOX
EAST

FIGURE 24. Preliminary Design of the BSR-Simulated Dosimetry Measurement Facility (SDMF). Neg 8010270-14

4. Surveillance Capsule Exposure Units and Values

R. L. Simons has tentatively completed the reanalysis of exposure units and values for 19 surveillance reports on Westinghouse PWRs, and the results are shown in Table 16. The newly calculated fluence values differ from those originally reported by an average upward shift of about 1.42 (~40%) with individual variations ranging from a low of 1.08 to a high of 2.11. The extent of the changes continues to indicate a need for re-evaluation of fluence values reported in other power reactor surveillance reports. In addition, future studies of ΔNDTT vs fluence will also require a reanalysis of the fluence values reported for test reactor irradiations.(47)

This HEDL Reactor Dosimetry Center work is the result of a recheck of 19 power reactor surveillance reports which are part of the bibliography of the MPC data base.(38) The magnitudes and frequency of the observed changes indicate that inconsistencies are nonnegligible and widespread. The HEDL work recalculated the fluence at the center of the surveillance capsule and was a test for the presence of inconsistencies in physics calculations, in calculated spectrum average cross-sections, in nuclear data, and in improper decay adjustments of reported foil activity adjusted for the known power time history of the reactor, as well as inconsistencies in the calculation of local flux perturbations due to the presence of the capsule itself. Corrections for photofission, thought to be small for the 19 Westinghouse type surveillance capsules investigated, were not considered in this first study.

Recently, workers at Westinghouse Pittsburgh have rechecked fluence values reported in existing power reactor surveillance reports and have reached conclusions similar to those given above. For instance, as a part of the Point Beach Unit No. 2 surveillance capsule "R" analysis,(31) S. Anderson has re-evaluated previously reported exposure values for capsules "V" and "T." The results of this reanalysis are shown in Table 17. The relative swing in the reported (before and after) capsule "V" and "T" results was 1.56. This swing was related to changes in the manner of handling, interpreting, and applying reported data and was not associated with changes in

TABLE 16

RE-EVALUATED EXPOSURE VALUES FOR SURVEILLANCE CAPSULES

Reactor	Unit	Capsule	old $\phi t > 1$	new $\phi t > 1$	Ratio new/old	dpa	Ratio dpa/ϕt new	D	A302B ΔTT Correlation Monitor (^{o}F)	time(sec)
Turkey Point	4	S	1.25+19	1.81+19	1.45	.032	1.77-21	2.96-10	115*	1.084+8
"	4	T	6.05+18	8.57+18	1.42	.0140	1.62-21	3.66-10	100	3.840+7
"	3	S	1.41+19	1.99+19	1.40	.0350	1.77-21	3.14-10	139	1.114+8
"	3	T	5.68+18	6.44+18	1.13	.0123	1.90-21	3.99-10	82	3.094+7
H. B. Robinson	2	S	3.2+18	5.09+18	1.59	.00936	1.83-21	2.14-10	80	4.385+7
"	2	V	4.51+18	7.76+18	1.72	.0137	1.77-21	1.29-10	105	1.061+8
Prairie Island	1	V	5.21+18	6.29+18	1.21	.0109	1.73-21	2.51-10	---	4.350+7
"	2	V	5.49+18	7.18+18	1.31	.0122	1.70-21	2.74-10	---	4.454+7
Point Beach	1	S	---	9.52+18	--	.0157	1.65-21	1.35-10	---	1.163+8
"	1	R	2.22+19	2.69+19	1.21	.0442	1.64-21	2.72-10	110	1.632+8
R. E. Ginnea	1	R	7.60+18	1.33+19	1.75	.0247	1.86-21	2.87-10	---	8.628+7
"	1	V	5.32+18	6.91+18	1.30	.0110	1.84-21	2.80-10	90	3.923+7
Kewaunee		V	5.59+18	7.23+18	1.29	.0123	1.71-21	2.96-10	---	4.167+7
San Onofre	1	A	1.20+19	1.48+19	1.23	.0310	2.04-21	5.30-10	115	5.846+7
"	1	D	2.36+19	4.98+19	2.11	.0887	1.78-21	9.95-10	150	8.914+7
Point Beach	2	T	9.45+18	1.02+19	1.08	.0170	1.67-21	1.55-10	105*	1.098+8
"	2	V	4.74+18	7.71+18	1.63	.0127	1.65-21	2.61-10	90*	4.859+7
Conn. Yank.		A	2.08+18	2.85+18	1.37	.00516	1.81-21	9.59-11	85	5.377+7
"		F	4.04+18	5.54+18	1.37	.00977	1.76-21	1.26.10	80	7.728+7
				Average	1.42					

*A533B Steel

TABLE 17

DOSIMETRY/PHYSICS RESULTS*

	CAPSULE V(1)	CAPSULE T(2)	CAPSULE R(3)
BEFORE ANALYSIS:	4.74×10^{18} (n/cm^2 > MeV)	9.45×10^{18}	20.1×10^{18}
AFTER REANALYSIS:	6.53×10^{18}	8.29×10^{18}	20.1×10^{18}
RATIO: AFTER/BEFORE	1.37	0.88	1.0
RELATIVE SWING	1.37/0.88 = 1.56		

E10.5 and E10.05.06 CONCLUSION: All surveillance reports and test reactor reports must be reanalyzed to improve exposure values!!!

1) Surveillance Capsule report, BMI, June 10, 1975
2) Surveillance Capsule report, West, August 1978
3) Surveillance Capsule Report, West, December 1979

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*EPRI Report by Yanichko, Anderson, Shogan, Lott, December 1979, Analysis of Capsule R from Point Beach #2 the measured reaction rates. CEN/SCK, EG&G ORTEC, HEDL, and RI will assist Westinghouse/EPRI in rechecking of the measured reaction rates for the capsule "R" Co-Al alloy (bare and cadmium covered), ^{237}Np(n,f), ^{238}U(n,f), ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,p)^{58}Co, and the ^{63}Cu(n,α)^{60}Co monitors. The iron monitor material is obtained from Charpy specimens. It is planned to have RI analyze some of the nonfission monitors for total helium content to assist in the verification of the radiometric monitors results. Since time history corrections are not required, HAFM monitors give a direct measurement of the fluence. Of particular interest will be a comparison of the measured and calculated Al(n,α) and Cu(n,α) total helium and ^{63}Cu(n,α)^{60}Co radiometric results.

The status of RI work related to the measurement of helium in pressure vessel steel chips to provide a direct means of obtaining three-dimensional information on the fast neutron fluence in operating PWR and BWR plants is discussed in the next section.

Based on the above studies, it is concluded that a rechecking of the fluence calculations in existing data bases would decrease the standard deviation in the statistically derived formulas giving Charpy-shift and other property changes as a function of fluence and chemistry. Further, the data bases should be improved by taking into account the different spectral shapes of the flux with the use of DPA as an exposure unit. This latter change will be most important when surveillance data and test reactor data are combined and subsequently used for ex-vessel structural component end-of-life predictions (see Table 3 of E706-80, reference 23).

5. Helium Accumulation Fluence Dosimetry for Reactor Vessel Surveillance

Several pieces of pressure vessel steel chips taken from a test specimen irradiated in the Belgian Reactor BR3 have been analyzed at Rockwell International for helium concentration. The purpose of the analysis was to see if the helium concentration could be used to provide a measurement of the fast neutron fluence received by the specimen. The test specimen was taken from the upper shell plate (Heat No. 19281) of the Yankee Rowe pressure vessel, and was irradiated about 15 years ago in a low flux surveillance position during the first BR3 core (so-called "first Westinghouse core"). The helium results from seven independent mass spectrometric analyses gave accurate results, despite the very low helium levels involved, providing a concentration of 0.692 ± 0.026 atomic parts per billion (6.92 x 10^{-10} atom fraction).

Recent calculations both at Rockwell and at the CEN/SCK Laboratories at Mol, Belgium, have attributed this helium concentration to the threshold Fe(n,α) reaction with high-energy fission neutrons, rather than to a thermal reaction with some possible boron impurity. If this is so, then helium measurements of BR3 pressure vessel scrapings would provide the fast fluence received by the BR3 vessel as a function of location of the scrapings. The same situation will apply to any vessel steels which have levels of boron sufficiently low that the thermal neutron (n,α) reactions with boron do not mask the fast neutron reactions with iron.

The next step in demonstrating if the helium accumulation technique can be used to retroactively determine the fast fluences received by existing pressure vessels is to determine the boron content of this and other pressure vessel steels. For the extremely low boron concentrations of interest here, this can only be accomplished by irradiating the steels in a highly thermalized flux (where only $^{10}B(n,\alpha)$ reactions will occur) and subsequently measuring the helium by mass spectrometry.

C. OTHER REACTOR DATA DEVELOPMENT AND TESTING STUDIES

1. Spectral Effects Associated with the Extrapolation of TMI-2 PWR Surveillance Capsule Measurements

As part of the NRC DSI (Division of Systems Integration) Radiation Embrittlement Program, and in support of the Reactor Vessel Materials Toughness Task Force (A-11), a review of the techniques used to infer pressure vessel damage from surveillance capsule measurements has been performed by J. Carew. This review concerned the effect of neutron spectral shifts, between surveillance capsule and vessel locations, on estimates of pressure vessel damage and the principal results of this review are as follows:

The extrapolation of surveillance capsule measurements to the pressure vessel 1/4T and 3T/4 locations is generally performed using fluence lead and attenuation factors. Typically, in the determination of these factors the shift in spectrum between the capsule and vessel locations is neglected. This assumption has been evaluated for the TMI-2 vessel/downcomer/shield configuration and found to result in ~40% under-prediction of ΔRT_{NDT} (change in reference nil ductility temperature) at the 3T/4 location for an accelerated surveillance capsule, and ~10% and ~50% under-prediction of ΔRT_{NDT} at the T/4 and 3T/4 locations, respectively, for a standard capsule.

2. Status of Exvessel Dosimetry in Arkansas Power and Light Company's Unit #1 PWR

The purpose of these exvessel experiments is to gain dosimetry and spectrum-unfolding experience in an RPV-related environment almost untouched by measurements. The previous (1979) annual report(2) provided analyses of dosimeters from the first irradiation, completed in February 1978. The following is a summary of the next three irradiations.

The second and third irradiation experiments were essentially a repeat of the first in an attempt to study reproducibility and the effects of different power histories, particularly during the last few days of irradiation, on dosimery results.

The fourth irradiation was the first time NP-237 dosimeters were included in the dosimetry sets. NP-237 is an extremely important dosimeter for flux in the 500 keV to 1 MeV energy range. The dosimery set containing the NP-237 was again at the reactor core centerline outside the pressure vessel. A second set of dosimeters was positioned in the cavity above core level in the vicinity of the nozzle penetration. This is believed to be the first time serious flux and spectrum measurements have been made on neutron leakage flux in this area of an operating PWR.

Dosimeters from the four experiments were counted in a cooperative program between the University of Arkansas (UA) and the Hanford Engineering and Development Laboratory. However, final analysis of the data has not yet been accomplished. The UA work is being supported by EPRI as well as the utility and more extensive cooperative measurements and analyses are planned.

3. Status of Metallurgical and Dosimetry Measurements in the Garigliano BWR, Rome, Italy

Charpy metallurgical specimens and associated radiometric neutron dosimeters were received at NBS from the Garigliano BWR Reactor in Rome, Italy. The metallurgy was shipped to R. Hawthorne, Naval Research Laboratory, for testing and the dosimeters will be sent to HEDL for analyses. Irradiation of this material ended August 8, 1979 after 129 equivalent full power days of operation. The materials were irradiated to approximately 4×10^{18} n/cm^2(E > 1 MeV) inside the pressure vessel of Garigliano.

The cooperative joint-research program with the Italian Atomic Power Authority (ENEL) in the area of reactor pressure vessel neutron dosimetry is conducted through F. Remondino, the ENEL representative in the USA, and with M. Galliani at the Garigliano Reactor, Rome, Italy. Improving and standardizing measurement methods for reactor pressure vessel dosimetry is now a recognized international activity and the early pilot efforts at Garigliano, at Unit #1 of Arkansas Power and Light, and at the BR-3 Reactor in Mol, Belgium have contributed significantly to this recognition.

4. Status of Dosimetry Measurements for the NRC HSST Program

Recommended values for the neutron flux (E>1 MeV) at various experimental locations within the dosimetry capsule simulating metallurgical capsules A and B are given in Tables 18 and 19. These results were obtained from the dosimetry experiment of the Fourth HSST Irradiation Series. The values are given for a BSR core power of 2 MW, and are estimated to be accurate to within ±5% (1σ) or better.

The measurements have been performed by means of the two transportable NaI(Tl) spectrometers calibrated at the CEN/SCK Mol, Cavity ^{235}U Thermal Fission Neutron Spectrum Standard Field. The methods and techniques are the same as applied to the neutron spectral characterization of the PCA Pressure Vessel Simulator. The good accuracy obtained stems largely from the use of this "Benchmark Field Referencing" approach. The directly observed experimental quantities, gathered in Tables 20 and 21, are the equivalent fission fluxes for each selected dosimetry reaction in capsules A and B.

It has been assumed that the neutron spectrum along the axis of a given vertical experimental channel does not change with the distance to the capsule top. As shown by Tables 22 and 23 this is true within experimental uncertainties, except very close to the edge (location 2.04 in. in center channel).

Figs. 25 and 26 are plots of the relative axial variation of the fast flux as a function of distance to the capsule top. Complete documentation of this work, including an analysis of the uncertainties, will be prepared for future HSST progress report dissemination.

TABLE 18. Neutron Fluxes (E>1 MeV) in BSR-HSST Capsule Mock-Up "A", at a BSR Core Power of 2.0 MW. Data derived from equivalent fission flux measurements and transport theory calculations.

Experimental Location[a]	$^{103}Rh(n,n')^{103m}Rh$	$^{115}In(n,n')^{115m}In$	$^{58}Ni(n,p)^{58}Co$	$^{27}Al(n,\alpha)^{27}Na$	Recommended (Average)
South Channel					
16.5	1.94(12)[b]	2.17(12)	2.09(12)	2.17(12)	2.09(12)
11.0	2.11	2.32	2.19	2.28	2.23
5.5	1.59	1.77	1.68	1.74	1.70
Center Channel					
13.04	1.12	1.25	1.23	1.25	1.21
7.54	.993	1.08	1.08	1.13	1.07
2.04	.601	.713	.661	.715	.673
East Channel					
16.5	.988	1.02	1.13	1.21	1.09
11.0	1.04	1.28	1.20	1.27	1.20
5.5	.812	.831	.907	.972	.881

[a] Locations are in inches, measured from the top of the dummy capsule.

[b] Read 1.94×10^{12} neutrons $cm^{-2}\ s^{-1}$.

TABLE 19. Neutron Fluxes (E>1 MeV) in BSR-HSST Capsule Mock-up "B", at a BSR Core Power of 2.0 MW. Data derived from equivalent fission flux measurements and transport theory calculations.

Experimental Location[a]	^{103}Rh(n,n')^{103m}Rh	^{115}In(n,n')^{115m}In	^{58}Ni(n,p)^{58}Co	^{56}Fe(n,p)^{56}Mn	^{27}Al(n,α)^{24}Na	Recommended (Average)
South Channel						
16.5	1.63(12)[b]	1.84(12)	1.90(12)	2.21(12)	1.97(12)	1.91(12)
11.0	1.77	1.95	1.94	2.19	1.95	1.96
5.5	1.34	1.45	1.44	1.61	1.45	1.46
Center Channel						
16.5	.895	1.01	1.08	1.26	1.12	1.07
12.0	1.00	1.10	1.12	1.29	1.12	1.13
8.25	.884	.960	.966	1.10	.981	.978
4.5	.668	.730	.725	.857	.741	.744
2.375	.518	.549	.588	.657	.586	.580
0.25	.307	.336	.368	.458	.407	.375

[a] Locations are in inches, measured from the top of the dummy capsule.

[b] Read 1.63×10^{12} neutrons cm^{-2} s^{-1}.

TABLE 20. Experimental Equivalent Fission Fluxes in BSR-HSST Capsule Mock-up "A" at a BSR Core Power of 2.0 MW.

Experimental Location[a]	^{103}Rh(n,n´)^{103m}Rh	^{115}In(n,n´)^{115m}In	^{58}Ni(n,p)^{58}Co	^{27}Al(n,α)^{24}Na
South Channel				
16.5	3.56(12)[b]	2.80(12)[b]	1.67(12)[b]	1.53(12)[b]
11.0	3.86	3.00	1.75	1.61
5.5	2.92	2.28	1.34	1.23
Center Channel				
13.04	2.25	1.58	.839	.745
7.54	2.00	1.36	.737	.670
2.04	1.21	.901	.452	.425
East Channel				
16.5	1.97	1.29	.780	.733
11.0	2.08	1.62	.828	.772
5.5	1.62	1.05	.626	.589

---

[a] Locations are in inches, measured from the top of the dummy capsule.

[b] Read 3.56 × 10^{12} neutrons cm^{-2} s^{-1}; note, these four values would be the same only if the BSR-HSST capsule Mockup spectrum at the given location were the same as the ^{235}U fission spectrum.

TABLE 21. Experimental Equivalent Fission Fluxes in BSR-HSST Capsule Mock-up "B" at a BSR Core Power of 2.0 MW

Experimental Location[a]	^{103}Rh(n,n')^{103m}Rh	^{115}In(n,n')^{115m}In	^{58}Ni(n,p)^{58}Co	^{56}Fe(n,p)^{56}Mn	^{27}Al(n,α)^{24}Na
South Channel					
16.5	2.84(12)[b]	2.34(12)[b]	1.57(12)[b]	1.60(12)[b]	1.47(12)[b]
11.0	3.08	2.48	1.60	1.59	1.46
5.5	2.34	1.85	1.19	1.17	1.08
Center Channel					
16.5	1.70	1.25	.757	.764	.711
12.0	1.90	1.36	.783	.781	.715
8.25	1.68	1.19	.677	.665	.624
4.5	1.27	.905	.508	.519	.471
2.375	.985	.681	.412	.398	.373
0.25	.584	.417	.258	.277	.259
East Channel					
11.0	1.54	1.12	.653	.659	.610
5.5	1.00	.846	.492	.494	.469

---

[a] Locations are in inches, measured from the top of the dummy capsule.

[b] Read 2.84 × 10^{12} neutrons cm^{-2} s^{-1}; note, these five values would be the same only if the BSR-HSST Capsule Mock-up spectrum at a given location were the same as the ^{235}U fission spectrum.

READEX FICHE

TABLE 22. Integral Spectral Indices in BSR-HSST Capsule Mock-up "A"

Experimental Location[a]	^{103}Rh(n,n′)^{103m}Rh[b]	^{115}In(n,n′)^{115m}In	^{58}Ni(n,p)^{58}Co
	^{27}Al(n,α)^{24}Na	^{27}Al(n,α)^{24}Na	^{27}Al(n,α)^{24}Na
South Channel			
16.5	2.33	1.83	1.09
11.0	2.40	1.86	1.09
5.5	2.37	1.85	1.09
Theory[c]	2.60	1.83	1.14
Theory[d]	2.32	1.71	1.10
Center Channel			
13.04	3.02	2.12	1.13
7.54	2.98	2.03	1.10
2.04	2.85	2.12	1.06
Theory[c]	3.36	2.11	1.14
Theory[d]	2.75	1.85	1.08

[a] Locations are in inches, measured from the top of the dummy capsule.

[b] Spectral index or equivalent fission flux ratio.

[c] Dosimetry capsule calculation.

[d] Metallurgical capsule calculation.

TABLE 23. Integral Spectral Indices in BSR-HSST Capsule Mock-up "B"

Experimental Location[a]	$\frac{^{103}\mathrm{Rh}(n,n')^{103m}\mathrm{Rh}}{^{27}\mathrm{Al}(n,\alpha)^{24}\mathrm{Na}}$[b]	$\frac{^{115}\mathrm{In}(n,n')^{115m}\mathrm{In}}{^{27}\mathrm{Al}(n,n')^{24}\mathrm{Na}}$	$\frac{^{58}\mathrm{Ni}(n,p)^{58}\mathrm{Co}}{^{27}\mathrm{Al}(n,\alpha)^{24}\mathrm{Na}}$	$\frac{^{56}\mathrm{Fe}(n,p)^{56}\mathrm{Mn}}{^{27}\mathrm{Al}(n,\alpha)^{24}\mathrm{Na}}$
South Channel				
16.5	1.93	1.59	1.07	1.09
11.0	2.11	1.70	1.10	1.09
5.5	2.17	1.71	1.10	1.08
Theory[c]	2.33	1.70	1.10	0.97
Theory[d]	2.35	1.73	1.12	0.97
Center Channel				
16.5	2.39	1.76	1.06	1.07
12.0	2.66	1.90	1.10	1.09
8.25	2.69	1.91	1.08	1.07
4.5	2.70	1.92	1.08	1.10
2.375	2.64	1.83	1.10	1.07
0.25	2.25	1.61	1.00	1.07
Theory[c]	2.99	1.95	1.10	0.95
Theory[d]	2.78	1.86	1.09	0.95

[a] Locations are in inches, measured from the top of the dummy capsule.

[b] Spectral index or equivalent fission flux ratio.

[c] Dosimetry capsule calculation.

[d] Metallurgical capsule calculation.

ORNL-DWG 80-16268

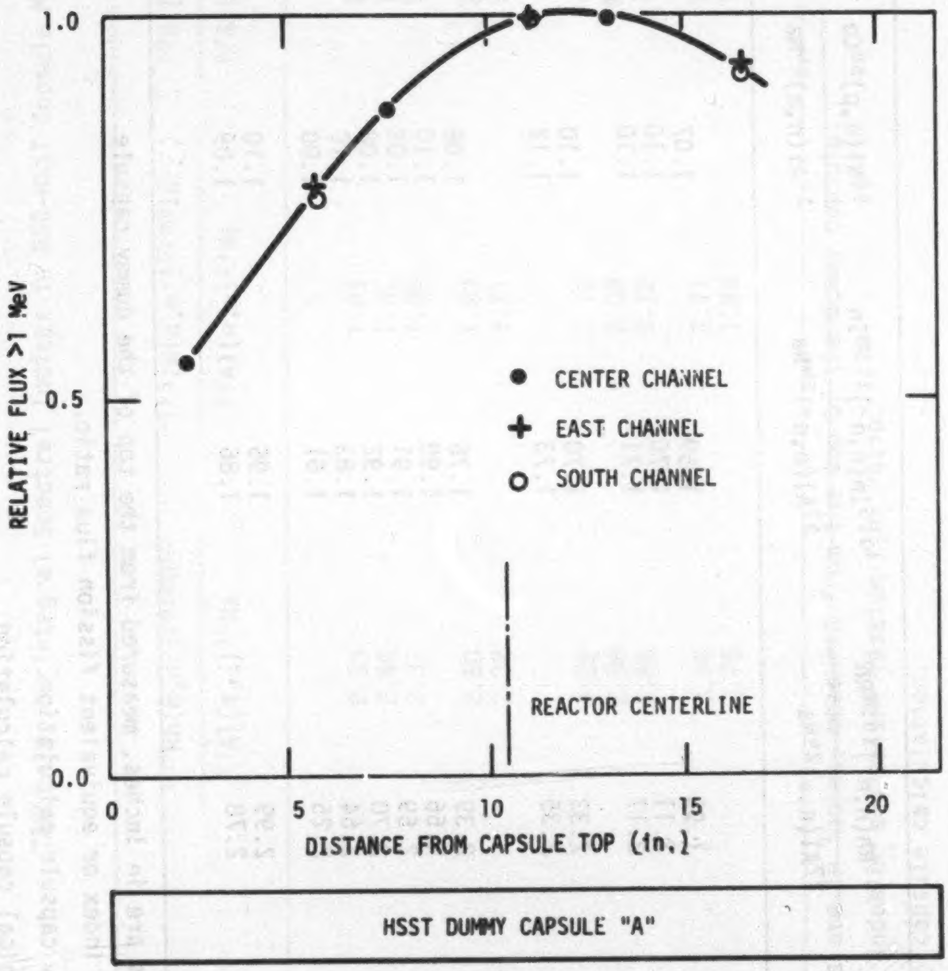

FIGURE 25. Relative Flux Distribution as a Function of the Distance from the Capsule Top, in the Dummy HSST Capsule at the BSR. Neg 8010270-13

ORNL-DWG 80-16269

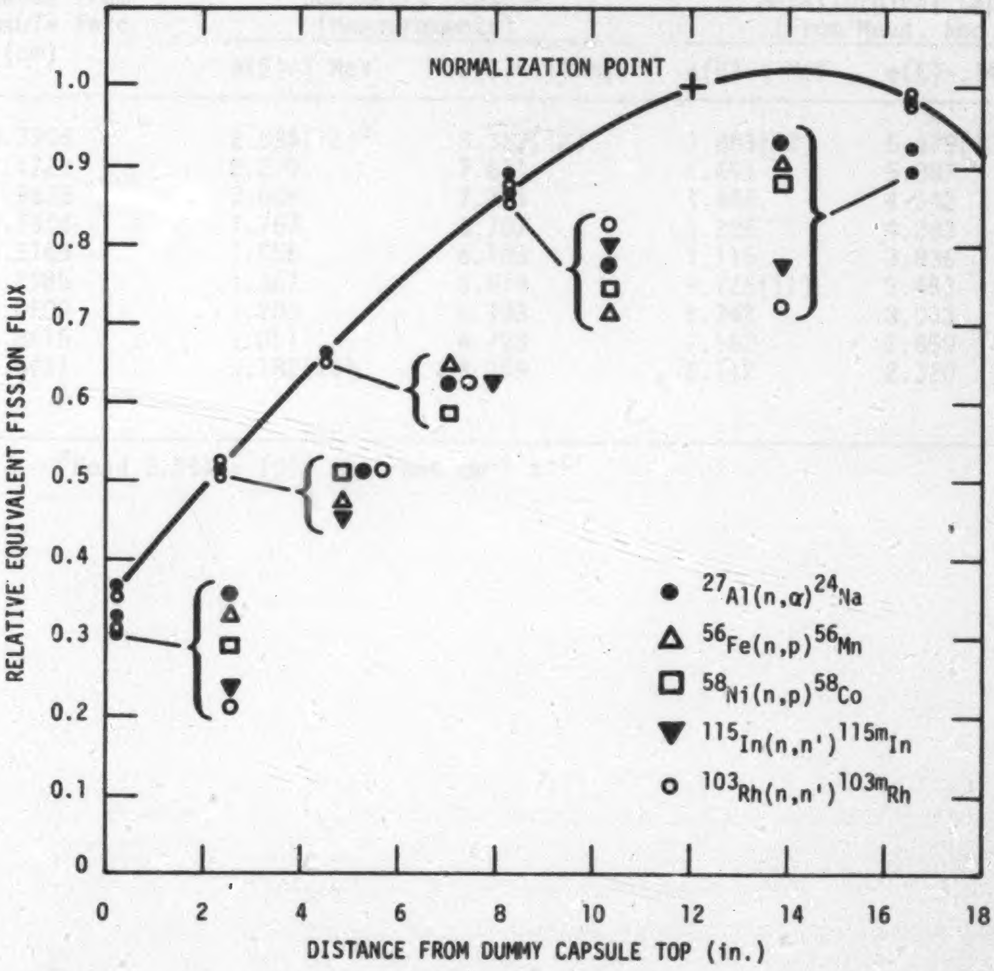

FIGURE 26. Relative Equivalent Fission Flux Along the Center Channel of the Dummy HSST Capsule, as a Function of the Distance from the Capsule Top. Neg 8010270-12

The results of the present measurements, after correction has been applied for the differences between the dosimetry mockup and the real metallurgical capsule, are given in Tables 24 and 25. These values can be combined with in-situ neutron fluence measurements to provide an improved neutronic characterization of this HSST capsule.

TABLE 24. Estimated Neutron Fluxes (E>1.0 MeV) in BSR-HSST Capsule "A" at a Core Power of 2 MW

Distance from Capsule Face (cm)	Dosimetry Capsule (Measurements)		Metallurgical Capsule (From Meas. and Calc.)	
	$\phi(E)>1$ MeV	$\phi(E)>.14$ MeV	$\phi(E)>1$ MeV	$\phi(E)>.14$ MeV
0.3908	2.594(12)[a]	8.382(12)	1.883(12)	5.479(12)
1.1723	2.279	7.811	1.651	5.087
1.9538	2.008	7.251	1.464	4.642
2.7354	1.767	6.707	1.285	4.283
3.5169	1.555	6.183	1.115	3.836
4.2985	1.367	5.679	9.726(11)	3.453
5.0800	1.200	5.193	8.342	3.033
5.8615	1.051	4.723	7.182	2.659
6.6431	9.182(11)	4.269	6.112	2.320

---

[a] Read 2.594×10^{12} neutrons $cm^{-2}\ s^{-1}$.

TABLE 25. Estimated Neutron Fluxes (E>1.0 MeV) in BSR-HSST Capsule "B" at a Core Power of 2 MW

Distance from Capsule Face (cm)	Dosimetry Capsule (Measurements)		Metallurgical Capsule (From Meas. and Calc.)	
	$\phi(E)>1$ MeV	$\phi(E)>.14$ MeV	$\phi(E)>1$ MeV	$\phi(E)>.14$ MeV
0.3908	2.371(12)[a]	6.886(12)	2.114(12)	6.007(12)
1.1723	2.094	6.439	1.842	5.589
1.9538	1.843	5.966	1.608	5.088
2.7354	1.622	5.516	1.402	4.604
3.5169	1.424	5.072	1.231	4.166
4.2984	1.249	4.650	1.057	3.726
5.0800	1.093	4.242	9.207(11)	3.327
5.8615	9.553(11)	3.855	7.849	2.888
6.6431	8.318	3.479	6.677	2.512

[a] Read 2.371 × 10^{12} neutrons cm^{-2} s^{-1}.

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*Available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and/or the National Technical Information Service, Springfield, Va. 22161

APPENDIX A

MASTER MATRIX FOR LWR PRESSURE VESSEL SURVEILLANCE STANDARDS - REVISION OF E706-79

For ASTM Committee E10 and ASTM Society Ballot

June 5, 1980

Notice

Working document for ASTM E10 Committee use only.
Not for publication except as expressly approved
by the Chairmen of E10, E10.05, and/or E10.02.

Publication for comment as a part of the NRC Annual
Report for the LWR-PV Surveillance Dosimetry Improvement
Program approved by the Chairmen of E10 and E10.05.

MASTER MATRIX FOR LWR PRESSURE VESSEL SURVEILLANCE STANDARDS[1]

1. SCOPE

1.1 This master matrix standard describes a series of standard practices, guides, and methods for the prediction of neutron induced changes in LWR Pressure Vessel (PV) steels throughout a pressure vessel's service life, Figures 1 and 2. Some of these are existing ASTM standards, some are ASTM standards that have been modified, and some are newly proposed ASTM standards. General requirements of content and consistency are discussed in Section 7. More detailed writers' and users' information, justification, and specific requirements for the five practices, six guides, and five methods are provided in Sections 3, 4, and 5. Applicable documents, references, and a Matrix Standard steering committee Task Group are discussed in Sections 2 and 6. The summary type information that is provided in Sections 3 and 4 is essential for establishing proper understanding and communications between the writers and users of this set of matrix standards. It cannot be easily, if at all, extracted from the applicable documents and references by individual writers and users.

1.2 This document is intended as a reference and guide to the preparation, revision and use of standards in the series and for planning and scheduling purposes, particularly as may concern future activities of E10.05 and E10.02. This index is to help ensure the accomplishment of an objective irrespective of the time required, the number of ASTM committees concerned, or the complexity of the issues involved.

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(1) This method is under the jurisdiction of ASTM Committee E-10 on Nuclear Technology and Applications.

1.3 This master matrix standard provides a guide to ASTM standards related to the energy-critical areas that are to be developed under the category of Fission Reactor Development, Section 10, of E584, Recommended Guide for Developing the (ASTM) Voluntary Consensus Standards Required to Help Implement the National Energy Plan and E583, Recommended Practice for Systematizing the Development of (ASTM) Voluntary Consensus Standards for the Solution of Nuclear and Other Complex Problems.[2] With reference to Figure 1 of E584, Comprehensive Matrix to Identify the Areas of Need for Which (ASTM) Standards Will Be Required to Help Implement the National Energy Plan, this E706 Master Matrix standard is identified with the horizontal and vertical subdivisions, respectively:

ENERGY SOURCE-RESEARCH, DEVELOPMENT, AND APPLICATION AREA

- Fission Reactor Development
  Converter
  1) LWR Pressurized Water (PWR)
  2) LWR Boiling Water (BWR)

STANDARDS-NEED AREA

- Analysis, Design, and Construction
- Quality Assurance and Reliability

1.4 To account for neutron radiation damage in setting pressure-temperature limits and making fracture analyses (see Ref. 2-7 and 9-14) neutron-induced changes in reactor pressure vessel steel fracture toughness must be predicted, then checked by extrapolation of surveillance program data during a vessel's service life. Uncertainties in the predicting methodology can be significant. Techniques, variables, and uncertainties associated with the physical measurements of PV steel property changes are not considered in this master matrix, but elsewhere.(1)[3] The techniques, variables, and uncertainties related to (i) neutron dosimetry, (ii) physics (neutronic and gamma effects) and (iii) metallurgical damage correlation procedures and data are addressed in this master matrix.(2,3,4) The main variables of concern to (i), (ii), and (iii) are:

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(2) Annual Book of ASTM Standards, Part 45; with reference to the E583 matrix approach, Subcommittee E10.05 on Dosimetry assumes the lead responsibility for developing and maintaining this LWR Pressure Vessel Surveillance Master Matrix with the liaison assistance of Subcommittee E10.02 on Behavior and Use of Nuclear Metallic Materials in Nuclear Systems.

(3) Parenthesized numbers refer to the list of references appended to this standard.

a) Steel chemical composition and microstructure
b) Steel irradiation temperature
c) Power plant configurations and dimensions--from the core edge to surveillance positions and into the vessel and cavity walls
d) Core power distribution
e) Reactor operating history
f) Reactor physics computations
g) Selection of neutron exposure units
h) Dosimetry measurements
i) Neutron spectral effects
j) Neutron dose rate effects

1.5 As older LWR pressure vessels become more highly irradiated, the predictive capability for changes in toughness must improve. Since during a vessel's service life an increasing amount of information will be available from test reactor and power reactor surveillance programs, better procedures to evaluate and use this information can and must be developed. This master matrix, therefore, defines the current (1) scope, (2) areas of application, (3) interrelationships, and (4) status and time table of development, improvement, validation, and calibration for a series of sixteen ASTM standards, as shown in Figures 1 and 2.

2. APPLICABLE DOCUMENTS

2.1 ASTM MATRIX STANDARDS

E583 Recommended Practice for Systematizing the Development of (ASTM) Voluntary Consensus Standards for the Solution of Nuclear and Other Complex Problems.

E584 Recommended Guide for Developing the (ASTM) Voluntary Consensus Standards Required to Help Implement the National Energy Plan.

2.2 ASTM STANDARDS REFERENCED IN E185, STANDARD PRACTICE FOR CONDUCTING SURVEILLANCE TESTS FOR LIGHT-WATER COOLED NUCLEAR POWER REACTOR VESSELS.

2.2.1 ASTM Standards

A 370 Methods and Definitions for Mechanical Testing of Steel Products

E 8 Tension Testing of Metallic Materials

E 21 Recommended Practice for Elevated Temperature Tension Tests of Metallic Materials

E 23 Notched Bar Impact Testing of Metallic Materials

E 208 Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels

E 482 Recommended Practice for Neutron Dosimetry for Reactor Pressure Vessel Surveillance

E 560 Recommended Practice for Extrapolating Reactor Vessel Surveillance Dosimetry Results

2.2.2 American Society of Mechanical Engineers Standard: Boiler and Pressure Vessel Code, Sections III and XI.

2.3 RELEVANT OTHER KEY ASTM STANDARDS NOT REFERENCED IN E185

C 859 Definition of Terms Relating to Nuclear Materials

E 170 Definitions of Terms Relating to Dosimetry

E 261 Determining Neutron Flux, Fluence, and Spectra by Radioactivation Techniques

E 380 Standard for Metric Practice

E 521 Recommended Practice for Neutron Radiation Damage Simulation by Charged-Particle Irradiation

E 693 Practice for Characterizing Neutron Exposures in Ferritic Steels in Terms of Displacements Per Atom (DPA)

2.4 OTHER ASTM DOCUMENTS[4]

Standards on Terms and Statistical Methods of Committee E 11

Regulations Governing ASTM Technical Committees

Form and Style for ASTM Standards

2.5 NUCLEAR REGULATORY DOCUMENTS

Code of Federal Regulations, Chapter 10, Part 50, Appendix G, "Fracture Toughness Requirements." (10CFR Part 50, Appendix G)

Code of Federal Regulations, Chapter 10, Part 50, Appendix H, "Reactor Vessel Materials Surveillance Program Requirements." (10CFR Part 50, Appendix H)

Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage on Reactor Vessel Materials," U. S. Nuclear Regulatory Commission, April 1977.

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(4) Available from ASTM Headquarters, 1916 Race St., Philadelphia, PA 19103.

LWR-PV Irradiation Surveillance Improvement Program Progress Reports:

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NUREG/CR-0285, U.S. Nuclear Regulatory Commission
NUREG/CR-0550, U.S. Nuclear Regulatory Commission
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NUREG/CR-1240, Volume 1, U.S. Nuclear Regulatory Commission
" Volume 2, U.S. Nuclear Regulatory Commission
" Volume 3, U.S. Nuclear Regulatory Commission
NUREG/CR-1241, Volume 1, U.S. Nuclear Regulatory Commission
NUREG/CR-1747, U.S. Nuclear Regulatory Commission

2.6 WORLD LIST OF NUCLEAR POWER PLANTS

2.7 CHRONOLOGY OF PAST EVENTS AND REFERENCE DOCUMENTATION

Table 1 provides a chronological listing of important events, results, benefits, and references prior to the 1979 establishment of this Master Matrix for LWR Pressure Vessel Surveillance Standards. It is intended for use by authors and users of this set of 16 ASTM Standards; see Section 7.

3. LWR PRESSURE VESSEL SURVEILLANCE - JUSTIFICATION, REQUIREMENTS, AND STATUS OF WORK

3.1. Aging light water reactor pressure vessels (LWR-PV) are accumulating significant neutron fluence exposures, with consequent changes in their state of steel embrittlement. The U.S. Nuclear Regulatory

Commission estimates that there are approximately 20 operating U.S. PWRs that will have beltline materials with marginal toughness, relative to the existing requirements of Appendices G and H of 10 CFR Part 50 and Regulatory Guide 1.99 sometime within their service life. Recognizing that accurate and validated measurement methods are needed to periodically evaluate the metallurgical condition of these reactor vessels, international multilaboratory work directed toward the improvement of LWR-PV surveillance is in progress.(2,3,4) The primary concern here is to improve and standardize surveillance tests, neutron dosimetry, damage correlation, and the associated reactor analysis procedures and data used for predicting the integrated effect of neutron exposure to LWR pressure vessels and support structures.

3.2 Immediate objectives of the international multilaboratory work are (1) to establish updated and improved surveillance tests, neutron dosimetry, damage correlation, and the associated reactor analysis procedures and data in ASTM standards for LWR-PV surveillance programs, and (2) to perform supporting validation and calibration experiments in benchmark neutron fields, reactor test regions, and operating power reactor surveillance positions. The goal of this activity is to establish consistent and accurate procedures and data as well as to guide the acquisition, reporting, and documentation of the required neutron field characterization formation that is used to correlate irradiation effects information and predict end-of-life (EOL) changes in PV steels and support structures.

3.3 The assessment of the radiation-induced degradation of material properties in a power reactor pressure vessel requires characterization of the neutron field from the edge of the reactor core to boundaries outside of the pressure vessel. Measurements of neutron flux, fluence, and spectrum for this characterization are associated with two distinct components of LWR-PV irradiation surveillance procedures: (1) proper calculational estimates of the neutron fluence delivered to in-vessel surveillance positions, various locations in the vessel wall, and ex-vessel support structures and surveillance positions and (2) understanding the interrelationship between material property changes in reactor vessels, in vessel support structures, and in metallurgical test specimens irradiated in test reactors and at accelerated neutron flux positions near the pressure vessel in operating power reactions; see Sections 4 and 5.

3.4 The first component requires validation and calibration in a variety of neutron irradiation test facilities, including LWR-PV mockups, power reactor surveillance positions, and the related benchmark neutron fields, Tables 2 and 3. The benchmarks also serve as a permanent measurement reference for neutron flux and fluence detection techniques, which are continually under development, and widely applied by laboratories with different levels of capability. The second surveillance procedure component requires a serious extrapolation of neutron-induced mechanical property change data obtained from test reactors and power reactor surveillance positions to locations inside the body of the pressure vessel and inside of ex-vessel support structures, Table 4. The neutron flux at the vessel inner wall is up to one order of magnitude lower than at surveillance specimen positions and up to two orders of magnitude lower than for test reactor positions. At the vessel outer wall, the neutron flux is one order of magnitude or more lower than at the vessel inner wall. Furthermore, the neutron spectrum at, within, and leaving the vessel is substantially altered; see Tables 3 and 5.

3.5 In order to meet the LWR-PV radiation monitoring requirements, a variety of neutron flux, fluence, and damage detectors are employed, most of which are passive; see Table 2. Each detector must be validated for application to the higher flux and harder neutron spectrum of the test reactor test regions and to the lower flux and degraded neutron spectrum of the surveillance positions; see

Tables 3 and 5. Required detectors must respond to neutrons of various energies, so that multigroup spectra can be determined with accuracy sufficient for adequate damage response estimated for PV and support structure steels at EOL; see Figure 3.

3.6 The necessity for well established and documented test reactor and pressure vessel mockup facilities for dosimetry and physics investigations and for irradiation of metallurgical specimens was recognized early. High flux [Oak Ridge Research Reactor-Poolside Facility (ORR-PSF), Table 3] and low flux [Pool Critical Assemblies (PCA), Table 2] versions of pressure vessel mockups have been established.(2) As specialized benchmarks, these facilities will provide well-characterized neutron environments where active and passive neutron dosimetry, various types of LWR-PV neutron field physics calculations, and temperature-controlled metallurgical damage exposures are brought together for validation and calibration. The neutron radiation field characteristics for surveillance capsule in- and ex-vessel power reactor positions will be simulated in these mockup facilities; see Tables 3 and 5.

3.7 The necessity for a few selected operating PWR and BWR power reactor benchmark facilities for testing, validation, and calibration of physics computational methods, processing and adjustment codes, nuclear data, and dosimetry techniques were also recognized; see Tables 2 and 3.(2,5)

3.8 The results of the measurement and calculational strategies outlined here are being made available for use by the nuclear industry as ASTM standards. Federal Regulation 10CFR50 already calls for adherence to several ASTM standards which require establishment of a surveillance program for each power reactor and incorporation of flux monitors for post-irradiation neutron field evaluation. Revised and new standards in preparation will be carefully structured to be up-to-date, flexible, and above all, consistent;

see Sections 6 and 7. The existing ASTM standards have been referenced and those in preparation or to be updated are described in Section 5.

4. SIGNIFICANCE AND USE

4.1 MASTER MATRIX

This matrix document is written as a reference and guide to the use of existing as well as unwritten standards to help manage in the development and application of standards needed for LWR-PV Surveillance Programs. It is written in accordance with the generic matrix approach advocated in ASTM Standards E583 and E584. Sections 4.2, 4.3, 4.4, and 4.5 are provided to assist the authors and users involved in the preparation, revision, and application of these standards; see Section 7.

4.2 APPROACH AND PRIMARY OBJECTIVES

4.2.1 Improved and standardized procedures and reference data are recommended in regard to (i) dosimetry, (ii) physics (neutronics and gamma effects), and (iii) metallurgical damage correlation methods and data associated with the analysis, interpretation, and use of nuclear reactor test and surveillance results.(2-20)

4.2.2 Existing state-of-the-art practices associated with (i), (ii), and (iii), if uniformly and consistently applied, can provide reliable (10 to 30%, 1σ) estimates of changes in LWR-PV steel fracture toughness during a reactor's service life.

4.2.3 Existing conservatism or nonconservatism associated with the variables (Section 1.4) related to (i), (ii), and (iii) must be reduced by improved practices and subsequent documentation and reporting of surveillance program results.

4.2.4 Application of improved practices and more complete documentation and reporting of test and power reactor results is essential to develop improved metallurgical data bases for reference standards, such as Reg. Guide 1.99 and Section III of the ASME Boiler and Pressure Vessel Code, Part NF2121, which requires that the materials used in reactor pressure vessels supports "...shall be made of materials that are not injuriously affected by ...irradiation conditions to which the item will be subjected."(7)

4.2.5 By the use of this series of standards and the uniform and consistent documentation and reporting of estimated changes in LWR-PV steel facture toughness at the 10 to 30% (1σ) confidence level, the nuclear industry and licensing and regulatory agencies can continue to establish realistic LWR power plant operating conditions and limits, such as those now defined in Appendices G and H of 10CFR Part 50, Reg. Guide 1.99, and the ASME Boiler and Pressure Vessel Code.

4.2.6 The uniform and consistent application of this series of standards will allow the nuclear industry and licensing and regulatory agencies to properly administer their responsibilities in regard to LWR power reactors that may develop materials with marginal toughness relative to existing and future requirements of Appendices G and H of 10CFR Part 50, Reg. Guide 1.99, and the ASME Boiler and Pressure Vessel Code.

4.3 DOSIMETRY ANALYSIS AND INTERPRETATION(2,5,6,9,11,12,15)

When properly implemented, validated, and calibrated by vendor/utility groups, state-of-the-art dosimetry practices exist which are adequate for existing and future LWR power plant surveillance programs. The uncertainties and errors associated with the individual and combined effects of the different variables (a) through (j) of Section 1.4 are considered in this and Sections 4.4 and 4.5. In these sections, the accuracy (uncertainty and error) statements that are made are quantitative and representative of state-of-the-art technology, Table 1. Their correctness and use for making EOL predictions for any given LWR power plant, however, are dependent on such factors as (1) the existing plant surveillance program, (2) the plant geometrical configuration, and (3) available surveillance results from similar plants. For example, consider the error statements made in Section 4.3.2.1 about surveillance capsule detector analysis and interpretation; conservative and nonconservative means that if a surveillance capsule PV steel measured value of property change (such as ΔNDTT) were determined per unit fluence greater than 1 MeV, without correcting for perturbation effects, the value could be 10 to 20% too high if based on a single iron flux monitor result and 30 to 50% too low if based on a single fission flux monitor result, respectively. As emphasized in Section III-A of Reference 9, however, these effects are not unique and are dependent on (1) the surveillance capsule design, (2) the configuration of the reactor core and internals, and (3) the location of the surveillance capsule within the reactor geometry. Further, the statement that a result could be in error (conservatively or nonconservatively relative to an EOL prediction) is dependent on how the neutron and gamma ray fields are estimated for a given reactor power plant.(15) For most of the error statements in Sections 4.3, 4.4, and 4.5, it is assumed that these estimates are based on reactor transport theory calculations that have been normalized to the core power history, see Section 4.4.1.2, and not to surveillance capsule dosimetry results. If the latter had been the case, then the error effect of the individual detector perturbations might be negligible or, at least, considerably lessened.

The Section 4.3, 4.4, and 4.5 accuracy statements, consequently, are intended for use in helping the standards' writer and user to determine the relative importance of the different variables in regard to the application of the set of sixteen ASTM standards, Figure 1, for (1) LWR-PV surveillance programs, (2) as instruments of licensing and regulation, and (3) for establishing improved metallurgical data bases.

4.3.1 Required Accuracies and Benchmark Field Referencing

4.3.1.1 The accuracies (uncertainties and errors)[5] desirable for LWR-PV steel exposure definition are of the order of ±10-15% (1σ), while exposure accuracies in establishing trend curves should preferably not exceed ±10% (1σ). In order to achieve such goals, the response of neutron dosimeters should therefore also be interpretable to accuracies within ±10-15% (1σ) in terms of exposure units and be measurable to within ±5% (1σ).

4.3.1.2 Dosimetry "Inventories" should be established in support of the above for use by vendor/utility groups and research and development organizations.

4.3.1.3 Benchmark field referencing of research and utilities' vendor/service laboratories is in progress which is:

- Needed for quality control and certification of current and improved dosimetry practices;
- Extensively applied in standard and reference neutron fields, PCA, PSF, PWRs, BWRs, and a number of test reactors to quantify and minimize uncertainties and errors; see Tables 2 and 3.

4.3.2 Surveillance Capsule Dosimetry Detector Analysis and Interpretation

4.3.2.1 Result of Neglect of Flux Perturbations--conservative exposure values by 10 to 20% when based on iron monitors alone and nonconservative by 30 to 50% for fission monitors alone. Uncorrected combined results cause correlation discrepancies in the 50 to 70% range.(9)

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(5) Uncertainty in the sense treated here is a scientific characterization of the reliability of a measurement result and its statement is the necessary premise for using these results for applied investigations claiming high or at least stated accuracy. The term error will be reserved to denote a known deviation of the result from the quantity to be measured. Errors are usually taken into account by corrections.(8)

4.3.2.2 Result of Neglect of Photoreactions--nonconservative exposure values by 10 to 50% for fission and non-fission threshold monitors.(5)

4.3.2.3 Result of Neglect of Burn-In of Fissile Products for Threshold Fission Reactions--nonconservative exposure values by large percentage depending on fluence (≥10%).

4.3.2.4 Result of Neglect of Difference Between Surveillance Capsule Flux and Reactor Power Time Histories--conservative or nonconservative exposure values by 10 to 40%, depending on fuel loadings and reactor operations.

4.3.3 Impact of Neglect of Combined Detector Effects of 4.3.2

4.3.3.1 Nonconservative (up to perhaps 100%) and conservative (up to perhaps 50%) for estimates of changes in PV steel fracture toughness, depending on the reactor configuration and normalization procedures (4.4.1.2).

4.3.3.2 Completely unacceptable accuracy (a swing of up to 150%) for establishing data base exposure values used to establish Reg. Guide 1.99 and other trend curves.

4.3.4 Status of Benchmark Field Referencing Work for Dosimetry Detectors

4.3.4.1 PCA experiments with the simulated surveillance capsule and power reactor tests are providing initial data for studying these effects (4.3.2); the PCA/PSF perturbation experiments are providing more exact data for more realistic PWR and BWR power plant surveillance capsule configurations and will allow utilities' vendor/service laboratories to test, validate, calibrate, and correct their current practices.(2,6,9) The PSF surveillance capsule test will provide further data, but of a more one-dimensional nature. Further PCA experimentation together with some test reactor work will complete the benchmark field quantification of these effects. PWR and BWR tests in progress or planned will complete the power reactor validation/calibration studies; see Tables 2 and 3.(2,5,6,15)

4.3.5 Additional Validation Work for Dosimetry Detectors

4.3.5.1 Establishment of nuclear data, σ(E)s-ASTM/ENDF/B, photoreaction σ(E)s, etc. reference files.

4.3.5.2 Establishment of proper quality assurance procedures for sensor set designs and individual detectors.

4.3.5.3 Interlaboratory comparisons using standard and reference neutron fields, PCA, PSF, and other test reactors which provide adequate validations and calibrations; see Tables 2 and 3.

4.3.6 Impact of Validation Work Associated with 4.3.5

4.3.6.1 Elimination of utilities' vendor/service laboratories' "laboratory-to-laboratory" biases of 10 to 20% or more for individual and combined dosimetry detector results, including corrections for photoreactions, fission yields, burn-in, etc.

4.4 REACTOR PHYSICS ANALYSIS AND INTERPRETATION(2,5,6,9,15)

When properly implemented, validated, and calibrated by vendor/utility groups, state-of-the-art reactor physics practices exist which are adequate for in- and ex-vessel estimates of PV-steel changes in fracture toughness for existing and future power plant surveillance programs.

4.4.1 Required Accuracies and Benchmark Field Referencing

4.4.1.1 The accuracies desirable for LWR-PV steel (surveillance capsule specimens and vessels) exposure definition are of the order of ±10-15% (1σ). Under idealized environmental conditions (benchmarking), modern computational techniques are currently capable of predicting absolute in- and ex-vessel neutron exposures and reaction rates per unit reactor core power to within ±15% (but generally not to within ±5%). The accuracy will be quite less, however, in applications to actual power plants because of geometrical and other complexities.(5,6)

4.4.1.2 There are at least three ways in which calculated in- and ex-vessel neutron and gamma ray fields can be benchmarked for power reactors:(6)

- by transport theory calculations normalized to the core power history;
- by transport theory calculations normalized to surveillance dosimetry results;
- by transport theory calculations normalized to measurements in the cavity behind the vessel and its thermal insulator.

The current practice of separating these normalization procedures for these three approaches can allow up to approximately 100% nonconservative and up to approximately 50% conservative estimates of changes in PV steel fracture toughness using surveillance capsule data.

4.4.2 Power Plant Reactor Physics Analysis and Interpretation

4.4.2.1 Result of Neglect of Benchmarking--1/4T vessel wall (etc.) estimates of damage exposure are not currently evaluated versus experiment--current results may be conservative or nonconservative by at least 50% (1σ) or more. (The Table 1 IAEA calculational intercomparison of PWR PV damage rate calculations: "Reactor Shielding Benchmark No. 2," shows factors of 2 or more differences).(6) Further, current "Lead Factors," based on the different ways they can be calculated (fluence > 0.1 or > 1.0 MeV and dpa) may not always be conservative; i.e., some surveillance capsules have been positioned in-vessel such that the actual lead factor is very near unity--no lead at all. Also the differences between lead factors based on fluence E > .1 or > 1 MeV and dpa can be significant, perhaps 50% or more.(2,15)

4.4.3 PCA Blind Test

4.4.3.1 Test of transport theory methods under clean geometry and clean core source conditions.(2,6)

4.4.3.2 This is a necessary but not sufficient benchmark test of the adequacy of a vendor/utility group's power reactor physics computational tools.

4.4.3.3 The standards recommendation should be that the vendor/utility groups' observed differences between their own calculated and the PCA measured integral and differential exposure and reaction rate parameters be used to validate and improve their calculational tools (if the differences fall outside of the PCA experimental accuracy limits.

4.4.4 PWR and BWR Generic Power Reactor Tests

4.4.4.1 Test of transport theory methods under actual geometry and variable core source conditions.(2,5,6)

4.4.4.2 This is a necessary and partly sufficient benchmark test of the adequacy of a vendor/utility group's power reactor physics computational tools.

4.4.4.3 The standards recommendation should be that the vendor/utility groups' observed differences between their own calculated and the selected PWR or BWR measured integral and differential exposure and reaction rate parameters be used to validate and improve their calculational tools (if the differences fall outside of the selected PWR or BWR experimental accuracy limits).

4.4.4.4 The power reactor "benchmarks" that are being established for this purpose are identified and discussed in references(2,5,and 6) and their references. (Also see Tables 2 and 3.)

4.4.5 Operating Power Reactor Tests

4.4.5.1 This is both a necessary and sufficient test of transport theory methods under actual geometry and variable core source conditions, but for a particular type or class of vendor/utility group power reactors. Here actual in-vessel surveillance capsule and any required ex-vessel measured dosimetry information will be utilized as in 4.4.4 above.(2,5,6,15)

4.4.5.2 Accuracies associated with surveillance program reported values of exposures and reaction rates are expected to be in the 10 to 30% (1σ) range, depending on the number of plant years of experience.

4.5 METALLURGICAL DAMAGE CORRELATION ANALYSIS AND INTERPRETATION(2-7, 9-15)

When properly implemented, validated, and calibrated by vendor/utility groups, state-of-the-art metallurgical damage correlation practices exist which are adequate for in- and ex-vessel estimates of PV-steel changes in fracture toughness and embrittlement for existing and future power plant surveillance programs.

4.5.1 Required Accuracies and Benchmark Field Referencing

4.5.1.1 The accuracies desirable and achievable for LWR-PV steel (test reactor specimens, surveillance capsule specimens, and vessels and support structures) data correlation and data extrapolation (to predict fracture toughness and embrittlement changes both in space and time) are of the order of ±10-30% (1σ). In order to achieve such a goal, however, the metallurgical parameters (ΔNDTT, upper shelf, yield strength, etc.) must be interpretable to well within ±20 to 30% (1σ). This mandates that in addition to the dosimetry and physics variables already discussed, the individual uncertainties and errors associated with a number of other variables [neutron dose rate, neutron spectrum, irradiation temperature, steel chemical composition, and microstructure] must be minimized and results must be interpretable to within the same ±10 to 30% (1σ) range.

4.5.1.2 Advanced sensor sets (including dosimetry, temperature and damage correlation sensors) and practices are being established in support of the above for use by vendor/utility groups.(2,6,9)

4.5.1.3 Benchmark field referencing of utilities' vendor/service laboratories, as well as advanced, practices is in progress or being planned which is:(2,5,6,9,10,15)

- Needed for validation of data correlation procedures and time and space extrapolations (to PV positions: surface, 1/4T, etc.) of test reactor and power reactor surveillance capsule metallurgical and neutron exposure data.
- Being or will be tested in test reactor neutron fields to quantify and minimize uncertainties and errors--see Tables 2 and 3 (included here is the use of damage correlation materials--steel, quartz, etc.).

4.5.2 Surveillance Capsule Dosimetry and Metallurgical Data Analysis and Interpretation

4.5.2.1 Result of Neglect of Detector Effects (4.3.2)--Nonconservative (up to perhaps 100%) and conservative (up to perhaps 50%) for estimates of changes in PV steel fracture toughness [This does not include exposure unit difference, flux spectrum, or flux level effects; assumes all correlations are based on fluence >1 MeV.]

4.5.2.2 Result of Neglect of Flux Spectrum and Exposure Units--Using fluence >1 MeV, nonconservative (up to perhaps 20 to 60%) for 1/4T to 3/4T positions and nonconservative (up to perhaps 70%) for ex-vessel positions, depending on the reactor configuration (this assumes that dpa is a reasonable measure of actual residual damage), see Tables 3 and 5.(2)

4.5.2.3 Result of Neglect of Flux Level--Potentially conservative by large percentages.(10,12) Existence of effect has yet to be separately demonstrated; it must be shown not to be associated with uncertainties and errors in other variables.

4.5.2.4 Result of Neglect of Temperature--Nonconservative or conservative depending on melt wire results, thermal hydraulics and gamma heating (up to perhaps 40% or more).

4.5.2.5 Result of Neglect of Chemistry--Nonconservative, depending on the steel (up to factors of about 3 for Cu and perhaps up to 100% or more for other elements).

4.5.2.6 Result of Neglect of Microstructure--Nonconservative or conservative, depending on the steel (up to perhaps 50 to 100%).

4.5.3 Impact of Neglect of Combined Effects of 4.5.2

4.5.3.1 Conservative (up to perhaps a factor of 3) and nonconservative (up to perhaps a factor of 2) for prediction of changes in PV steel fracture toughness and embrittlement, depending on the reactor configuration and normalization procedures (4.4.1.2).

4.5.3.2 Completely unacceptable accuracy (a swing of a factor of 2 or more) for establishing data base reference standards such as Reg. Guide 1.99.

4.5.4 Benchmark Field Referencing--Status of Work

4.5.4.1 The PSF (all positions: surveillance, surface, 1/4T, 1/2T, and the void box, Table 3) will provide needed validation data on all variables--dosimetry, physics, and metallurgy.(2,6) New test reactor data will be available as well as new surveillance capsule results that have been better benchmarked by vendor/service laboratory/utility groups.(2,5,6,10,15)

4.5.5 Reg. Guide 1.99, MPC, and EPRI Data Bases

New MPC and EPRI[6] data bases will be studied by ASTM E10.02 and E10.05 Subcommittees, MPC, vendors, utilities, EPRI, and NRC

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(6) Materials Property Council (MPC) and the Electric Power Research Institute (EPRI).

contractors to establish improved data bases for existing test and power reactor measured property change data. ASTM task groups will recommend the use of updated and new exposure units (total fluence, fluence >0.1 MeV, fluence >1.0 MeV and dpa) for the MPC and EPRI data bases. The success of this effort will depend on good cooperation between research and individual service laboratories and vendor/utility groups. An ASTM dosimetry cross section file based on ENDF/B-V and corrections for all known variables (perturbations, photoreactions, spectrum, burn-in, yields, flux time history, etc.) will be used as required and justified. Test reactor data will be addressed in a similar manner, as appropriate.

5. MATRIX DESCRIPTION

The following index of ASTM Standard Practices, Guides, and Methods constitutes the Master Matrix, describes the scope of individual standards, and provides other relevant information for the series of sixteen LWR-PV surveillance standards.[7] Unless explicitly noted, all standards are new as opposed to updated or revised versions of existing standards. Figures 1 and 2 indicate by titles the area or areas of application and show the interrelationships and time table, respectively, for the development, revision, validation, and calibration for this series of ASTM Standards.[8]

5.1 SURVEILLANCE AND CORRELATION PRACTICES

5.1.1 Analysis and Interpretation of Nuclear Reactor Surveillance Results--IA (E10.05)[9]

5.1.1.1 Scope

This new Master Practice will describe the methods to be used in the analysis and interpretation of data obtained

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(7) For standards which are in the draft stage and have not been assigned an ASTM number, see Figures 1 and 2, the Master Practice, I-A, will be very explicit and provide necessary detailed information on the procedures and data that are expected to be recommended in unnumbered reference standards.

(8) Cross referencing of these standards is to be done by means of the designations given in Figure 1. Thus, the Analysis and Interpretation of Nuclear Reactor Surveillance Results practice should be referred to as E706(I-A).

(9) Indicates the ASTM E10 Subcommittee that has the primary responsibility for the preparation of the standard.

from LWR pressure vessel surveillance programs and based on the results of that analysis will establish a formula to be used to evaluate present and future conditions of LWR pressure vessels and support structures.

This practice will rely on and ties together the application of several other ASTM Standard Practices, Guides, and Methods which are in various stages of development; see Figures 1 and 2. Support subject areas which will be discussed include (1) reactor physics calculations, (2) dosimetry selection and analysis, and (3) exposure units and temperature correlation.

5.1.1.2 Discussion

This new IA ASTM Master Practice, Figure 1, will set out the best available procedures for reactor pressure vessel toughness and embrittlement predictions.(2-7, 9-15) It can be (1) referenced as an instrument of licensing and regulation and (2) used with IE for the establishment of improved metallurgical data bases. As shown in Figure 2, this practice will be developed, revised, tested, and validated over a period of several years. The analysis and interpretation steps expected to be contained in the Master Practice are outlined in Table 4. This Master Practice will be restricted to direct applications related to surveillance programs which are established in support of the operation, licensing, and regulation of LWR PWR and BWR nuclear power plants. Procedures and data related to the analysis, interpretation, and application of test reactor results will be addressed in ASTM Standards IE, Damage Correlation for Reactor Vessel Surveillance, and IC, Extrapolating Reactor Vessel Surveillance Results, Figures 1 and 2.

5.1.2 Surveillance Tests for Nuclear Reactor Vessels--IB (E10.02)

5.1.2.1 Scope

This ASTM E185 Practice covers procedures for monitoring the radiation-induced changes in the mechanical properties of ferritic materials in the beltline of light-water cooled nuclear power reactor vessels. This practice includes guidelines for designing a minimum surveillance program, selecting materials, and evaluating test results. This IB Practice, Figure 1, was developed for all light-water cooled nuclear power reactor vessels for which the predicted maximum neutron fluence (E>1 MeV) at the end of the design lifetime exceeds 1 x 10^{21} n/m^2 (1 x 10^{17}n/cm^2) at the inside surface of the reactor vessel.

5.1.2.2 Discussion

Predictions of neutron radiation effects on pressure vessel steels are considered in the design of light-water cooled nuclear power reactors. Changes in system operating parameters are made throughout the service life of the reactor vessel to account for radiation effects. Because of the variability in the behavior of reactor vessel steels, a surveillance program is warranted to monitor changes in the properties of actual vessel materials caused by long-term exposure to the neutron radiation and temperature environment of the given reactor vessel. This practice describes the criteria that should be considered in planning and implementing surveillance test programs and points out precautions that should be taken to ensure that: (1) capsule exposures can be related to beltline exposures, (2) materials selected for the surveillance program are samples of those materials most likely to limit the operation of the reactor vessel, and (3) the tests yield results useful for the evaluation of radiation effects on the reactor vessel.

The design of a surveillance program for a given reactor vessel must consider the existing body of data on similar materials in addition to the specific materials used for that reactor vessel. The amount of such data and the similarity of exposure conditions and material characteristics will determine their applicability for predicting the radiation effects. As a large amount of pertinent data becomes available it may be possible to reduce the surveillance effort for selected reactors by integrating their surveillance programs.

The planned revisions of this standard, Figure 2, will more completely address the combined use, need, and accuracy requirements of dosimetry, physics, metallurgical damage correlation and temperature measurement techniques for surveillance programs in support of other matrix standards.(2-7, 9-15)

5.1.3 Surveillance Test Results Extrapolation--IC (E10.05

5.1.3.1 Scope

The current ASTM E560 Practice is intended to be used to relate neutron flux density derived from monitors in surveillance capsules to the corresponding flux density at the inner diameter and within the pressure vessel wall of a light water-moderated reactor. It describes analytical and analytical-experimental approaches that can be used to determine the variation in neutron flux density and energy spectrum between a surveillance capsule location and the pressure vessel wall. Procedures for reporting the results of these analyses are also suggested. It does not cover the relationship of the neutron damage observed at the surveillance location to that occurring within the pressure vessel wall. Rather, it deals solely with the neutron flux density monitoring aspects of the surveillance program.

5.1.3.2 Discussion

This IC ASTM E560 Practice, Figure 1, will be revised following the schedule in Figure 2. The revised scope will include specification of procedures for the extrapolation and interpolation of dosimetry and metallurgical surveillance data from test reactors, pressure vessel benchmark mockups, and reactor accelerated surveillance locations to and within the pressure vessel wall and to ex-vessel positions.(2-7, 9-15) New areas to be discussed within the existing scope may include (1) surveillance capsule perturbation effects, (2) use of benchmark fields and benchmarked calculations such as the PCA Blind Test, (3) possible use of adjoint physics calculations to provide an uncertainty estimate on the accuracy of the resulting dpa calculations at surveillance positions and within the vessel wall and support structures, (4) the effect of uncertainties and errors in reactor dimensions, and (5) the effect of reactor power cycle-to-cycle shifts in core fuel loadings. Recommended reactor physics procedures and data will be addressed in a new ASTM Standard, Application of Neutron Transport Methods for Reactor Vessel Surveillance, IID of Figures 1 and 2.

5.1.4 Displaced Atoms (DPA) Exposure Unit--ID (E10.05)

5.1.4.1 Scope

This ASTM E693 Practice describes a standard procedure for characterizing neutron irradiations of iron (and ferritic steels) in terms of the exposure index displacements per atom (dpa). It is assumed that the displacement cross section for iron is an adequate approximation for any ferritic steel. The application of this practice requires knowledge of the total fluence and the neutron-flux spectrum.

5.1.4.2 Discussion

A pressure vessel surveillance program requires a methodology for relating radiation-induced changes in materials exposed in test reactors and accelerated surveillance locations to the condition of the pressure vessel and support structures. An important consideration is that the irradiation exposures be expressed in a unit that is physically related to the damage mechanism.(2-7, 9-15)

A primary source of neutron radiation damage in metals is the displacement of atoms from their normal lattice sites. Hence, an appropriate damage exposure index is the number of times, on the average, that an atom has been displaced during an irradiation. This can be expressed as the total number of displaced atoms per unit volume, per unit mass, or per atom of the material. Displacements per atom is the most common. The number of dpa associated with a particular irradiation depends on the amount of energy deposited in the material by the neutrons, hence, depends on the neutron spectrum and fluence.

No simple correspondence exists in general between dpa and a particular change in a material property. An appropriate starting point, however, for relative correlations of property changes produced in different neutron spectra is the dpa value associated with each environment; see Tables 3 and 5. That is, the dpa values themselves provide a spectrum-sensitive index that may be a useful correlation parameter, or some function of the dpa values may affect correlation. The correlation of radiation effects data will be considered in a new ASTM Standard, Damage Correlation, IE, Figure 1. The Figure 2 schedule for the revision of the ID ASTM DPA Practice is supportive of the schedule for the development, revision, testing, and validation of the new ASTM IE Standard.

5.1.5 Damage Correlation for Reactor Vessel Surveillance--IE (E10.05)

5.1.5.1 Scope

This new ASTM Practice will describe the physically based correlation model and the statistical methods recommended for use for the correlation of neutron-induced property change data obtained from metallurgical specimens irradiated in test reactors, pressure vessel benchmark mockups, and LWR power plant surveillance capsules. The main variables of concern will be: (1) alloy chemistry, (2) metallurgical state, (3) temperature, (4) neutron flux level, (5) neutron spectrum, and (6) exposure units (total fluence, fluence >0.1 MeV, fluence >1.0 MeV, and dpa; see Section 1.4.

This practice will rely on the application of several other ASTM Standard Practices, Guides, and Methods which are in various stages of development. Further, dosimetry and physics methods to be used in the analysis, interpretation, and application of test reactor and pressure vessel benchmark mockup results will be addressed in this practice since they will not be considered in ASTM Practice IA, as stated in Section 5.1.1.2.

5.1.5.2 Discussion

One of the current objectives of LWR research and development programs is the verification of the quality of reference metallurgical data bases (Reg. Guide 1.99, MPC, EPRI, and others) used for LWR pressure vessel and support structure steel EOL predictions.(2-7, 9-15) Associated with this is the correlation of variations in "chemistry" and LWR "high flux level" test reactor and "low flux level" power reactor surveillance data, as shown in Figure 3.(2,9-12)

This new ASTM Practice will deal with the specification of procedures for effecting the best correlation between available neutron-induced metallurgical property change and neutron environmental characterization data (flux-fluence spectra and damage exposure parameters). Consideration will be given to the separate and combined effects of irradiation and metallurgical variables, Section 1.4, on changes in pressure vessel and support structure steel fracture toughness. This is ASTM Practice IE of Figure 1. The schedule for its development, revision, testing, and validation is shown in Figure 2. Areas which are expected to be included in the discussion and recommendations are: (1) saturation and annealing effects, (2) rate effects, (3) synergistic effects between copper and other elements such as nickel, (4) statistical evidence for the values of the parameters in recommended trend curves, (5) recommended statistical methods for correlating and handling metallurgical test reactor, pressure vessel benchmark mockup, and power plant surveillance capsule test results.

Dosimetry and physics (neutron and gamma) environmental characterization methods for test reactor irradiations will be specified that are consistent and supportive of those recommended in ASTM Standard Practice IA, Section 5.1.1, for operating LWR BWR and PWR power plant surveillance programs. This will allow test reactor research and power plant surveillance program results to be correlated with consistent and reliable exposure values.

5.2 SUPPORTING METHODOLOGY GUIDES

5.2.1 Application of Neutron Spectrum Unfolding Methods--IIA (E10.05)

5.2.1.1 Scope

This new ASTM Guide will describe the procedures and codes recommended for use for the determination of neutron flux-fluence spectra from multiple sensor measurements. The procedures described here will be restricted to test reactor and power reactor measurements for light water reactors. The applicable range of neutron energies is

0-18 MeV, provided appropriate detector response functions and input trial spectra (from physics calculations) are available. This Guide will address the uncertainties and errors associated with derived integral neutron field characterization and exposure parameters (total flux-fluence, flux-fluence >0.1 and >1.0 MeV, and dpa). This IIA Guide will rely on the application of several other ASTM Guides (IIB, IIC, IID, and IIE) and Methods (IIIA, IIIB, IIIC, and IIID) which are in various stages of development, Figures 1 and 2.

5.2.1.2 Discussion

The use of test reactor and power reactor surveillance results for the prediction of EOL pressure vessel and support structure steel changes in fracture toughness and embrittlement requires the measurement and determination of neutron flux-fluence spectra for neutron energies in the range of 0 to 18 MeV. For neutron energies below about 1.0 MeV, the information is needed for the assessment of the effect of lower energy neutrons on the interpretation and application of multiple sensor measurements; see Tables 3 and 5. That is, (1) for the adjustment of reactor physics results in the thermal, 1/E, and 0.01 to 1.0 MeV transition range, (2) for the determination of exposure values (total flux-fluence, flux-fluence >0.1 and >1.0 MeV, and dpa), and (3) for making corrections for target and product burn-in and burn-out effects for individual sensors and sensor covers (cadmium and gadolinium).

5.2.2 Application of ENDF/A Cross Section and Uncertainty Files--IIB (E10.05)

5.2.2.1 Scope

This new ASTM Guide will cover the establishment and use of an ASTM ENDF/A cross section and uncertainty/error tape file for (1) the analysis of single or multiple sensor measurements in LWR neutron fields and

(2) the calculation of spectral averaged damage cross sections for iron, quartz, and other sensors that might be used as damage exposure monitors. The neutron fields include surveillance positions in operating power reactors, test reactor regions, and benchmark neutron fields. This Guide will describe requirements for the tape file, including data format, individual cross-section evaluations and adjustments, and uncertainty/error estimates. The tape will be available from the National Nuclear Data Center (NNDC), Brookhaven National Laboratory.

5.2.2.2 Discussion

This new IIB ASTM Guide will be directly related to and should be used in conjunction with the new IIA Guide, "Application of Neutron Spectrum Unfolding Methods," Figure 1. It will be restricted to INTEND Code[10] processed 620 group averaged cross section values that are available from the NNDC for the ENDF/B Dosimetry File.(16) Uncertainties and errors will be specified for the 620 groups, including suggestions for assigning covariances between the groups. This information is required for the newer unfolding methods applied to the determination of flux-fluence spectra in four larger energy groups: (1) Thermal, (2) 1/E (thermal up to about 0.01 MeV, (3) a transition region between about 0.01 to 1 MeV, and (4) a fast region above about 1 MeV. The schedule for its development, revision, testing, and validation is shown in Figure 2.

5.2.3 Sensor Set Design and Irradiation for Reactor Surveillance--IIC (E10.05)

5.2.3.1 Scope

This new ASTM Guide will cover the selection, design, irradiation, and postirradiation handling of radiometric

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(10) This is an NNDC-developed code for processing the ENDF/B energy-dependent point-wise dosimetry cross-section file.

monitors (RM), solid state track recorders (SSTR), helium accumulation fluence monitors (HAFM), damage monitors (DM), and temperature monitors (TM) sensors and sensor sets. It will include the consideration of sensor and sensor set placement, sensor set covers (thermal neutron shields), target and product burn-in and burn-out effects, photoreaction effects, quality control of constituents, mass assay, and sensor and sensor set perturbations of the irradiation and the thermal temperature environments. Its use will be restricted to test reactor and power reactor measurements for light water reactors. This guide will rely on the application of several other ASTM Methods (IIIA, IIIB, IIIC, IIID, and IIIE), which are in various stages of development; see Figures 1 and 2.

5.2.3.2 Discussion

Characterization of the irradiation (neutron and gamma) and thermal (temperature) environments for LWR test reactor and operating BWR and PWR power plants is essential for the proper analysis, interpretation, and application of research and power reactor test and surveillance program results. Reactor physics (neutron and gamma) and thermal hydraulics (neutron and gamma heating) computations must be validated and calibrated by active and passive sensor and sensor set measurements.(2-6, 9-12, 15) This new IIC ASTM Guide will be used, therefore, as a reference in other ASTM standards, Figure 1, when passive sensor and sensor set measurements are recommended for LWR test and power reactor environmental characterization measurements. The schedule for its development, revision, testing, and validation is shown in Figure 2.

5.2.4 Application of Neutron Transport Methods for Reactor Vessel Surveillance--IID (E10.05)

5.2.4.1 Scope

This revised ASTM E482 Guide will describe the methodology for performing radiation transport calculations to determine the neutron and gamma flux spectra within LWR research and power reactors. These calculations are required as a basis for the correlation of research and power reactor results and subsequent prediction of the EOL fracture toughness and embrittlement of LWR pressure vessel and structural steel components. Consequently, the procedures will be developed to be applicable for design and operation calculations and for the evaluation and extrapolation of research and surveillance program measurement results obtained in operating reactors. The accuracy of reactor physics calculations will be considered together with benchmarking procedures for validating and calibrating the results of computations; see Section 4.4.

5.2.4.2 Discussion

This IID ASTM Guide, Figure 1, will address the radiation transport methods, data, results, and the uncertainties and errors associated with LWR calculated integral radiation field characterization and exposure parameters, such as total neutron flux-fluence, flux-fluence >0.1 and >1.0 MeV, and dpa. This Guide will be used as a reference in other ASTM Standards, Figure 1, when reactor physics (neutron and gamma) computations are recommended for LWR test and power reactor environmental characterization. The schedule for its development, revision, testing, and validation is shown in Figure 2.

5.2.5 Benchmark Testing of Reactor Vessel Dosimetry--IIE (E10.05)

5.2.5.1 Scope

This new ASTM Guide will describe and provide reference information on (1) the application of well-characterized neutron fields to the characterization of integral neutron sensors and the validation of techniques used to measure neutron flux, fluence, and fission rates, (2) the appropriate formulation of sensor reaction probabilities, (3) the procedures for flux transfer and spectral index calibration and measurement, and (4) a compendium of benchmark neutron fields.

5.2.5.2 Discussion

This guide will describe methodology for using the known characteristics of certain neutron fields to perform calibrations of fast neutron sensors, or to corroborate or validate procedures used to derive neutron field information from measurements of their response.(2,5,6,9,15) This is ASTM Guide IIE of Figure 1. The schedule for its development, revision, testing, and validation is shown in Figure 2.

The section dealing with the application of methods will have special emphasis on neutron dosimetry performed for the surveillance of the fast neutron exposure of pressure vessels in light water reactors. The development and characterization of controlled environments and test regions for benchmark referencing PV surveillance dosimetry will be discussed, including the development and maintenance of a compendium of reference information on these special benchmark neutron fields. Particular sections will discuss:

(1) The use of recognized standard neutron fields for the calibration of individual neutron sensors, including radiometric monitors (RM), solid state track recorders (SSTR), helium accumulation fluence monitors (HAFM), fission chambers, etc.

(2) The production and use of certified-neutron-fluence counting standards to determine interlaboratory consistency in measuring fast-neutron fluxes and fluences.

(3) The calibration and measurement of spectral indexes; i.e., spectrum-averaged cross section or response function ratios that are the fundamental integral parameters for assessing the characteristics of neutron energy spectra.

(4) Calibration and flux transfer measurements; i.e., procedures to measure fluxes in environments under investigation in terms of sensor response per unit neutron flux in a benchmark neutron field.

(5) Measurements which lessen the dependency of spectral characterizations on the uncertainties in the absolute magnitudes of neutron cross sections.

(6) The documentation of characteristics and associated uncertainties of standard fields and basic parameters derived from them.

(7) The formulation of reaction probability parameters and variances in terms of experimental parameters, cross sections and benchmark field calibrations.

5.2.6 Correlation of ΔNDTT with Fluence--IIF (E10.02)

5.2.6.1 Scope

This new ASTM Guide will describe the metallurgical data base, the curve-fitting techniques, and the resulting property change versus exposure curves (currently, only for fluence >1 MeV) for the Charpy shift in brittle ductile transition temperature for LWR pressure vessel materials. The main variables of concern will be: (1) steel types--material product form (plate, forging, weldment or HAZ material) and material specification (A533B, A302B, or A508), (2) neutron irradiation temperature range--approximately 277 to 310°C (530-590°F), (3) neutron exposure units and values--currently restricted to fluences >1 MeV in the 1×10^{16} to 10^{20} n/cm^2 range. This IIF ASTM Guide, Figure 1, will rely on the application of several other ASTM Standard Practices, Guides, and Methods which are in various stages of development. It is expected that three sets of property change versus exposure curves will be developed based on (1) a total metallurgical data base, (2) a test reactor experimental metallurgical data base, and (3) a power reactor surveillance capsule metallurgical data base. The selection of the appropriate set of curves for a given application will be left to the discretion of the user. Furthermore, for any given BWR or PWR power plant, these curves will be intended for use, primarily, for predictions required before plant specific surveillance data are available.

5.2.6.2 Discussion

Commercial reactor vessels are required to have a surveillance program to monitor neutron-induced changes in fracture toughness of the materials used in the construction of the reactor pressure vessel; see Section 1. The current practice is to estimate the fracture toughness using Charpy toughness data.(3,4, 10-14) To assure conservative operational margins for nuclear power plants, accepted and accurate interpretations of Charpy V-notch toughness results are therefore necessary. The Materials Properties Council (MPC), Inc., has recommended that the MPC Subcommittee 6 Report(17) on the "Prediction of the Shift in the Brittle-Ductile Transition Temperature of LWR Pressure Vessel Materials," be evaluated by ASTM Committee E10 with respect to the development of a new ASTM standard. Such a standard is identified as ASTM Guide IIF of Figure 1. The schedule for its development, revision, testing, and validation is shown in Figure 2. In addition to the MPC Report's metallurgical data base, and depending on availability, consideration will also be given to other metallurgical data and data bases, such as that being developed by EPRI,(10-12) in the preparation of this II-F Standard.

5.3 SENSOR MEASUREMENT METHODS

5.3.1 Analysis of Radiometric Monitors for Reactor Vessel Surveillance--IIIA (RM) (E10.05)

5.3.1.1 Scope and Discussion

This new ASTM Method will describe the use of Radiometric Monitors (RM) for neutron dosimetry in LWR applications. Measurement procedures will be specified for RM sensors by means of gamma ray or x-ray emission rates; determination of reaction rates including systematic effects such as interference from competing reactions, pulse losses, monitor selfshielding, branching ratios and fission yields. The useful flux-fluence range and limitations for the application of RM for LWR reaction and reaction rate measurements will be specified. Procedures for the calculation of RM reactions and reaction rates from radioactivity measurements and the irradiation time history will be included. The assessment and discussion of methods and techniques for estimating uncertainties and errors will be an important part of this standard; see Sections 7.2, 7.3, and 7.4. For example, the standard will address: (1) benchmark calibrations and (2) the use of multiple reaction products (^{140}Ba, ^{103}Ru, ^{95}Zr, ^{137}Cs, etc.) from fission monitors for the assessment of individual sensor time history corrections. This IIIA ASTM Method, Figure 1, will be referenced and used in conjunction with the new IIC ASTM Guide on "Sensor Set Design." Its use will be restricted to test reactor and power reactor measurements for light water reactors. The schedule for its development, revision, testing, and validation is shown in Figure 2.

5.3.2 Analysis of Solid State Track Recorder Monitors for Reactor Vessel Surveillance--III-B (E10.05)

5.3.2.1 Scope and Discussion

This new ASTM Method will describe the use of Solid State Track Recorders (SSTR) for neutron dosimetry in LWR applications. Measurement procedures will be specified for SSTR sensors by means of optical (and later more advanced) track counting techniques.(19) The useful flux-fluence range and limitations for the application of SSTR for LWR time-integrated reaction rate measurements will be specified. SSTR are truly passive fluence detectors that provide a permanent record for a dosimetry experiment without the need for time-dependent corrections, such as decay factors which arise with RM detectors. The assessment and discussion of methods and techniques for estimating uncertainties and errors will be an important part of this standard; see Sections 7.2, 7.3, and 7.4. This IIIB ASTM Method, Figure 1, will be referenced and used in conjunction with the new IIC ASTM Guide on "Sensor Set Design." Its use will be restricted to test reactor and power reactor measurements for light water reactors. The schedule for its development, revision, testing, and validation is shown in Figure 2.

5.3.3 Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance--IIIC (E10.05)

5.3.3.1 Scope and Discussion

This new ASTM Method will describe the use of Helium Accumulation Fluence Monitors (HAFM) for neutron dosimetry in LWR applications. Measurement procedures will be specified for HAFM sensors by means of high-sensitivity helium gas mass spectrometry.(20) The useful flux-fluence range and limitations for the application of HAFM for LWR time-integrated reaction rate measurements will be specified. HAFM, like SSTR, are truly passive fluence detectors that provide time-integrated reaction rate measurements without the need for time-dependent corrections, such as decay factors which arise with RM detectors. The assessment and discussion of methods and techniques for estimating uncertainties and errors will be an important part of this standard; see Sections 7.2, 7.3, and 7.4. This IIIC ASTM Method, Figure 1, will be referenced and used in conjunction with the new IIC ASTM Guide on "Sensor Set Design." Its use will be restricted to test reactor and power reactor measurements for light water reactors. The schedule for its development, revision, testing, and validation is shown in Figure 2.

5.3.4 Analysis of Damage Monitors for Reactor Vessel Surveillance--IIID (E10.05)

5.3.4.1 Scope and Discussion

This new ASTM Method will describe the use of Damage Monitors (DM) for dosimetry and damage correlation in LWR applications. Measurement procedures will be specified for different types of DM sensors. The useful flux-fluence-temperature range and limitations for the application of DM will be specified. DM sensors are important because they have a measured integral response for all neutrons that contribute to neutron damage in LWR PV and support structure steel materials and components. More particularly, they provide a measured response in the neutron energy range between about 0.01 to about 0.5 MeV, where RM, SSTR, and HAFM sensors have little or no response. The assessment and discussion of methods and techniques for estimating uncertainties and errors will be an important part of this standard; see Sections 7.2, 7.3, and 7.4. This IIID ASTM Method, Figure 1, will be referenced and used in conjunction with the new IIC ASTM Guide on "Sensor Set Design." Its use will be restricted to test reactor and power reactor measurements for light water reactors. The schedule for its development, revision, testing, and validation is shown in Figure 2.

Areas which may be covered in this new IIID Method are (1) use and interpretation of quartz and sapphire damage monitors, (2) saturation and annealing effects, (3) irradiation temperature effects, (4) dpa cross-section tables for quartz and sapphire, (5) use of well characterized standard reference heats of PV steels as damage monitors, (6) use of graphite and tungsten for damage monitor purposes, (7) the combined use of damage monitor and temperature information in deducing (at least qualitatively) the time at temperature, (8) the extent to which damage monitors can contribute quantitatively to increased accuracy in determining unfolded spectra, and (9) the use of damage monitors to avoid spurious results.

5.3.5 Analysis of Temperature Monitors for Reactor Vessel Surveillance--IIIE (E10.02)

5.3.5.1 Scope and Discussion

This new ASTM Method will describe the application of different types of Temperature Monitors (TM) and the use and analysis of measured data and derived information. The assessment and discussion of methods and techniques for estimating uncertainties and errors will be an important part of this standard; see Sections 7.2, 7.3, and 7.4. This IIIE ASTM Method, Figure 1, will be referenced and used in conjunction with the new IIC ASTM Guide on "Sensor Set Design." Its use will be restricted to test reactor and power reactor measurements for light water reactors. The schedule for its development, revision, testing, and validation is shown in Figure 2.

Major emphasis is expected to be on (1) the available melt wire alloys, (2) the importance of thermal contact between the monitor and the object or environment whose temperature is desired, (3) the encapsulation techniques, (4) the use of power time history curves in the interpretation of results, (5) the effect of gamma heating on thermal sensors, and (6) possible errors due to heat release in transmutation processes.

6. ESTABLISHMENT OF A MATRIX STANDARD TASK GROUP

6.1 An E10.05.04 Task Group of Subcommittee E10.05 has been formed for the specific purpose of acting as a steering committee on this set of ASTM Standards for LWR-PV Surveillance Programs. The primary function of this task group is to review the proposed ASTM Standards at various stages of preparation and revision to see that they are well related and mutually consistent, and that as a group they cover LWR-PV program elements and goals as set forth in Sections 3 and 4. In support of this activity are a number of other ASTM E10.02 and E10.05 Task Groups which address specific areas of concern. They are not listed separately here, but are identified in other ASTM publications.

7. GENERAL REQUIREMENTS OF CONTENT AND CONSISTENCY

7.1 Comments on these standards, their revision, and use should be considered at three levels. The first two levels represent necessary committee functions; the third includes remarks about details that may help the authors and users.

7.1.1 General Content

Topics, issues, and data that need to be covered; see Sections 1, 3, 4, and 5.

7.1.2 Organization and Consistency with Other Standards and References

Parallelism in content, overlapping with other standards, consistency in terminology, symbols, definitions, etc. Reference should be made to the appropriate Section 2, Table 1, and other applicable documents.

7.1.3 Editorial Suggestions

Form and Style for ASTM Standards; see Section 2.4.

7.2 STANDARDS SHOULD IDENTIFY AND DISCUSS ALL ELEMENTS OF ACCURACY

Analysis and measurement accuracies (uncertainties, errors, and correlations) in the areas of concern for this set of Matrix Standards may be difficult to determine or estimate as discussed in Section 4. Difficult or not, they should be properly addressed in each standard.(8) When uncertainties, errors, and correlations are well identified, as in integral reaction rate measurements, for example, they should be estimated and summarized in an accuracy table.(2) For more difficult kinds of uncertainties, errors, and correlations (e.g. in spectrum unfolding) a state-of-art analysis will have to be chosen and a statement will have to be included which indicates what the uncertainty, error, and correlation estimates do and do not cover. It will be necessary to accept incomplete or nonrigorous uncertainty, error, and correlation estimates when there is no readily available alternative. This is necessary because it is a basic program purpose to improve the existing situation regarding accuracy.(2-7, 9-15) In this regard, an ASTM E10.05.01 Task Group on "Uncertainty Analysis" has been established to assist in this aspect of the preparation and revision of the set of 16 ASTM Standards.

7.3 STANDARDS SHOULD BE COMPLETE WITH REGARD TO MEASUREMENT AND INTERPRETATION ISSUES.

All elements of the measurement and analysis issues should be identified and treated to the extent present capabilities allow. The Table 1 and Section 2 references should be used and referenced, as appropriate. Problems should not be ignored because there is no proper answer yet. Semiquantitative, advisory, or cautionary statements are often used in ASTM Standards. Field perturbations and photoreactions are two examples of measurement problems that are not as yet well understood and benchmarked, but must be considered.(2,5,6,15)

7.4 STANDARDS CONCERNED WITH SENSOR MEASUREMENT, DATA INTERPRETATION OR TRANSPORT CALCULATION SHOULD REFER TO BENCHMARK FIELD CALIBRATION AND VALIDATION.

The dosimetry improvement effort centers around benchmark calibration and validation of measurement and calculational tools--see Sections 3 and 4. This calibration alternative and/or validation requirement should be entered into the ASTM standards. For the sensor methods, a suggested statement which may be inserted in the section dealing with the derivation of a single reaction rate, is as follows: "Alternatively, a calibration of the composite neutron detection efficiency for the EXXX-XX Method may be obtained by exposure in a benchmark neutron field where the fluence and spectrum-averaged cross section are known." This specifically uses the term "calibration" for single detectors. It is proposed to use the term "validation," as distinct from "calibration," when multiple-sensors, cross sections, or the entire unfolding procedures are subject to a benchmark neutron field check.

7.5 STANDARDS SHOULD USE CONSISTENT TERMINOLOGY AND UNITS FOR NEUTRON FIELD QUANTITIES AND NUCLEAR PARAMETERS

Reference is made to ASTM C859, E-170, and E-380. The first entry for a Benchmark Field Compendium, to be distributed in 1980, will establish reaction rate formulations, cross sections, and computed and observed field quantities which will be employed to describe the benchmarks. Action will be taken to consider such definitions as candidates for inclusion in future revisions of E-170.

TABLE 1

CHRONOLOGICAL LISTING OF IMPORTANT EVENTS, RESULTS, BENEFITS, AND REFERENCES

DATE - EVENT - REFERENCE	RESULT AND/OR BENEFIT
JUNE 1961 - Publication of E185-61, ASTM Standard Recommended Practice for "Surveillance Tests for Nuclear Reactor Vessels."	• First U.S. Standard for LWR Pressure Vessel Surveillance
MAY 1971 - Interlaboratory Reaction Rate (ILRR) Program Established.	• Start of Coordinated Interlaboratory Integral Measurements in Selected Standard and Reference Benchmark Neutron Fields. • Validation/Calibration of Integral and Differential Dosimetry Data for Fission Reactors.
OCT 1972 - IAEA Specialist's Meetings on Radiation Damage Units for Ferritic and Stainless Steels [Proceedings of Seattle Meeting, Nucl. Eng. Design, Vol. 33, 1975].	• First Serious International Attempt to Standardize Exposure Units for Materials Irradiations. Calculational Model Adopted and Engineering Damage Cross-Sections Defined for Ferritic Steels.
Sept 1973 - IAEA Consultant's Meeting on Nuclear Data for Reactor Dosimetry [INDC (NDS) 56/U, 1973].	• International Agreement on Need to Develop Reference Dosimetry Files for LWR, FBR, and MFR Programs and Direction Established for Future Interlaboratory Program Work.
DEC 1973 - Publication of Recommended Procedures for Displacement Damage Calculations for U.S. FBR Program [HEDL-TME 73-76, 1973].	• Procedures Widely Adopted un U.S. FBR Program, With Carry-Over to LWR and MFR Programs.
Sept 1974 - Cross-Section Evaluation Working Group (CSEWG) Dosimetry File Task Force Meeting.	• Formal Identification of Dosimetry, Gas Production, Etc. Cross-Section Requirements. Specific Needs Established for ENDF/B for LWR, FBR, and MFR Programs.
DEC 1974 - ASTM E10 Mini-Symposium on "Neutron Dosimetry and Spectrum Analysis for Materials Applications" [Journal of Testing and Evaluation, Vol. 3, 1975].	• Early Attempt to Encourage Improved Understanding and Interaction between Dosimetrists and Metallurgists.
FEB 1975 - ILRR Program Results Published in Special Issue of Nuclear Technology Nuclear Technology,[Volume 25, 1975].	• Important Initial Step Completed in Validation/Calibration of Integral and Differential Dosimetry Data for Fission Reactors. Preliminary Data Testing Completed of Evaluated Dosimetry Cross-Sections on ENDF/B-IV and Sand II Files.
SEPT 1975 - First ASTM-EURATOM International Symposium on Reactor Dosimetry [EUR 5667 elf, Parts I and II, 1977].	• First Major World-Wide Effort to Identify and Assess Dosimetry Accuracies for LWR, FBR, and MFR Programs and Establish Improved Interlaboratory Cooperative Work.
OCT 1976 - CSEWG-ENDF/B Special Applications File (Safe) Subcommittee Established.	• Established Following CSEWG Task Groups: . Dosimetry File . Gas Production File . Activation File . Radioactive Decay File . Users Needs and Test Recommendations • Major Milestone - Important Step Towards International Standardization and Usage of Consistent Set of ENDF/B Data Files.
OCT 1976 - Joint IAEA-OECD (NEA) Technical Committee Meeting, Includes IAEA Calculational Intercomparison of PWR PV Damage Rate Calculations: Reactor Shielding Benchmark No. 2 [IAEA -]	• Identified Computational Problem Areas (Factor of Two or More Differences) Requiring Future Resolution.
NOV 1976 - IAEA Consultant's Meeting on Integral Cross-Section Measurements in Standard Neutron Fields [IAEA-208, 1978].	• First Major Step to Establish International Agreement on Interlaboratory Effort of Benchmark Field Studies. • LWR, FBR, and MFR Program Needs Identified, Reviewed and Updated.
NOV 1976 - IAEA Specialist's Meeting on Radiation Damage Units [Proceedings of Harwell Meeting, IAEA Conference-761146, 1977].	• Identified Two Specific Displacement Cross-Section Sets for Fission Reactor Structural Metals Based on Damage Model Recommended in October 1972. • Recommended That One Set be Adopted, Based on Single Set of Evaluated Reaction Cross-Sections, Applicable to Both Fission and Fusion Reactors.
MAR 1977 - Interlaboratory LWR Pressure Vessel Surveillance Dosimetry Program Established.	• Start of Coordinated Interlaboratory Effort to Prepare New and Updated ASTM Standards and Validate and Calibrate Them in Selected "Benchmark Neutron Fields" and Reactor "Test and Surveillance Regions." • New and Updated Validated/Calibrated ASTM Standards Expected to be Established in the period 1979 to 1984 for LWR Fission Reactors.

Cont'd

TABLE 1 (Cont'd)

DATE - EVENT - REFERENCE	RESULT AND/OR BENEFIT
MAR 1977 - CSEWG Dosimetry Task Group Meeting.	• Data Testing of ENDF/B-IV Dosimetry File. • Direction for Establishment of ENDF/B-V Dosimetry File.
MAR 1977 - International Specialist's Symposium on Neutron Standards and Applications [NBS Special Publication 493, 1977].	• Assessment of Progress in Neutron Standards and Applications. • Direction Established for Future Work for LWR, FBR, and MFR Programs.
MAY 1977 - Symposium on Nuclear Reactor Data in the Energy Range 10-40 MeV [BNL-NCS-50681, 1977].	• Assessment of Status of Neutron Cross-Sections from 10-40 MeV for MFR Program High Flux Neutron Sources. • Nuclear Data Needs and Research Direction Established for MFRs, with Carry-Over to LWR and FBR Programs.
OCT 1977 - Second ASTM-EURATOM International Symposium on Reactor Dosimetry [NUREG/CP-0004, Volumes 1 to 3, 1978].	• Major Step to Further Establish Working International Laboratory Cooperative Efforts for LWR, FBR, and MFR Programs. • Major Step Towards Implementation of International Standardization and Better Coordination of Joint Efforts of Dosimetrists, Metallurgists, and Reactor Analysts.
OCT 1977 - IAEA Specialist's Meeting at ORNL on Neutron Spectrum Unfolding [Current Status of Neutron Spectrum Unfolding, IAEA-TECDOC-221, 1979].	• Major Step to Establish International Agreement on Use of More Standardized Procedures of Determining Flux/Fluence Spectra. • Guidance for Preparation of ASMT Multiple Sensor Standard.
NOV 1977 - ASTM E10 Mini-Symposium on "Reactor Pressure Vessel Surveillance Programs: Recent Experiences and Future Directions."	• Important Step to Inform Metallurgists and Dosimetrists on State-of-the-Art Practices in Use for LWR-PV Surveillance Programs.
AUG 1978 - RSIC Workshop on "Theory and Application of Sensitivity and Uncertainty Analysis" [ORNL/RSIC-42, 1979].	• State-of-the-Art Review of Sensitivity and Uncertainty Analysis for Fast, Thermal, and Fusion Reactor Applications.
NOV 1978 - IAEA Advisory Group Meeting on "Nuclear Data for Reactor Dosimetry" [INDC(NDS)-103/M, 1978].	• IAEA Agreed to Establish and Maintain an International Reactor Dosimetry File (IRDF).
NOV 1978 - Publication of Parts I and II of E521-77, ASTM Standard Recommended Practice for "Neutron Radiation Damage Simulation by Charged Particle Irradiation." Part III on Correlation of Damage Produced by Neutrons is being developed.	• First U.S. Standard for Using and Correlating Charged Particle Irradiation Effects Data, Including DPA Exposure Units.
DEC 1978 - ASTM E10 Mini-Symposium on "Interfaces Between Neutron Dosimetry and Metallurgical Properties in Nuclear Pressure Vessel Surveillance."	• First Major Step to Encourage Improved Understanding and Interactions Between Dosimetrists, Metallurgists, and Reactor Analysts for LWR Pressure Vessel Surveillance Programs.
FEB 1979 - IAEA Specialists' Meeting on "Irradiation Embrittlement, Thermal Annealing and Surveillance of Reactor Pressure Vessels" [Proceedings to be Published as IAEA Document, 1980].	• Second Major Step to Encourage Improved Understanding and Interactions Between Dosimetrists, Metallurgists, and Reactor Analysts for LWR Pressure Vessel Surveillance Programs.
SEPT 1979 - IAEA Specialists' Meeting on "Accuracies in Correlation Between Property Change and Exposure Data from Reactor Pressure Vessel Steel Irradiations" [Proceedings to be published as IAEA Document, 1980].	• Third Major Step to Encourage Improved Understanding and Interactions Between Dosimetrists, Metallurgists, and Reactor Analysts for LWR Pressure Vessel Surveillance Programs.
OCT 1979 - Third ASTM-EURATOM International Symposium on Reactor Dosimetry [To be published as an Euratom Document, 1980].	• Major Step to Further Establish Working International Interlaboratory Cooperative Efforts for LWR, FBR, and MFR Programs. • Major Step Towards Implementation of International Standardization and Better Coordination of Joint Efforts of Dosimetrists, Metallurgists, and Reactor Analysts.
1970 to 1979 - ASTM International Symposia on Radiation Effects in Materials and ASTM Special Technical Publications (STPs).	• Reports on State-of-the-Art Metallurgy and Dosimetry. • Continuing Steps to Encourage Improved Understanding and Interactions Between Dosimetrists, Metallurgists, and Reactor Analysts.

TABLE 2

LIST OF NEUTRON FIELDS FOR LWR-PV DOSIMETRY VALIDATION AND CALIBRATION STUDIES

Neutron Field	Type of Dosimetry*	Status
Benchmarks for Calibration		
• ^{252}Cf and ^{235}U Fission Neutron Irradiation Facilities at NBS (Standard)	Sensor calibrations	Preparation of fluence counting standards in progress for participating laboratories. ^{252}Cf is absolute fast neutron fluence standard.
• ^{235}U Fission Field at CEN/SCK (Standard)	Sensor calibrations	Applied routinely to RM, DM and other sensor calibrations for PCA and other neutron field measurements.
• ISNF Irradiation Facilities at NBS (Standard)	Sensor calibrations (particularly detectors with response range below 0.5 MeV)	Fission rate ratios established with fission chambers. Application to RM and other sensors planned.
• CRMF Irradiation Facility at EG&G (Reference)	Sensor calibrations	Planning stage.
• PCA/PSF Irradiation Facilities at ORNL (Reference)	Sensor validations and surveillance perturbation investigations	BSR-SDMF being established at ORNL.
Benchmarks for Transport Calculation Validation		
• Iron Shells Field at CEN/SCK (Reference)	Active and passive RM, SSTR, NE	CEN/SCK spectrometry in progress. NBS fission chamber measurements completed. Planning stage for passive sensors.
• PCA at ORNL (Reference)	Active and passive RM, SSTR, NE	Radiometric, fission chamber, track recorder, and neutron and gamma spectrometry measurements in progress. Results used for "Blind Test" program.
Test Regions for Dosimetry Method** Validation		
• BSR-HSST Dosimetry-Test (Test Reactor at ORNL)	Radiometric monitors (RM)	Irradiation and dosimetry counting completed. Analysis of Interlaboratory dosimetry in progress.
• FRJ-1 and FRJ-2 (Test Reactor at KFA, Germany)	RM, HAFM, TM (Melt Wires), DM (Quartz)	"Test Region" irradiations in metallurgical rigs in progress. Interlaboratory dosimetry analysis in progress.
• DIDO, PLUTO, and HERALD (Test Reactors in UK)	RM, HAFM, TM, DM	Planning stage, UK results are available.
• Buffalo NRL-NRC (Test Reactor at University of Buffalo)	Radiometric monitors	Planning stage, NRL results are available.
• University of Virginia (Test Reactor)	Radiometric monitors	Planning stage, Westinghouse results are available.
• BR-3 (PWR Power Reactor at CEN/SCK, Mol, Belgium)	RM, HAFM, DM (Quartz), TM (Melt Wires)	In- and Ex-Vessel "Test Region" irradiations in progress.
• Arkansas Power and Light Company PWR Unit #1 (Russelville, Arkansas)	Radiometric monitors	Ex-vessel "Test Region" irradiations and sensor counting and analysis in progress.
• Garigliano Reactor (BWR Power Reactor at Rome, Italy)	Radiometric monitors	In-vessel "Test Region" irradiation completed and analysis in progress.
• Brown's Ferry 3 (BWR)	RM, SSTR, Ionization Chambers	In- and ex-vessel "Test Region" irradiations completed and analysis in progress.
• McGuire I (PWR)	RM, SSTR, Proton Recoil	Ex-vessel "Test Region" irradiations planned and scheduled.

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*RM: Radiometric Monitors
SSTR: Solid State Track Recorders
HAFM: Helium Accumulation Fluence Monitors
NE: Nuclear Emulsions
DM: Damage Monitors
TM: Temperature Monitors
BSR-SDMF: ORNL Bulk Shielding Reactor - Surveillance Dosimetry Measurement Facility.

**These neutron fields are also being used for physics and metallurgical damage correlation procedures and data validation and calibration studies.

TABLE 3

METALLURGICAL STEEL IRRADIATIONS - TEST FACILITIES ESTIMATED ENVIRONMENTAL PARAMETER VALUES FOR ASSESSING TEMPERATURE, FLUX LEVEL, AND SPECTRAL EFFECTS

TEST FACILITY	IRRADIATION TEMPERATURE (°C)	$\left[\frac{\bar{\sigma}_{DPA}}{\phi(>1.0MeV)}\right]$ (BARNS)	$\left[\frac{\overline{\sigma}\ Fe^{54}(n,p)}{\phi(>1.0MeV)}\right]$ (MILLI-BARNS)	$\frac{\phi(>0.1MeV)}{\phi(>1.0MeV)}$	FLUX (>1.0MeV) LEVEL (n/cm -sec)
BELGIUM					
BR2					
• DGR POSITION (CORE)	150°C	$1390^{c}(\quad)^{d}$	87^{b}	2.1	$3x10^{14}$
BR3					
• SURVEILLANCE POSITION	260^{f}	$1422^{c}(\quad)^{d}$	120^{b}	1.9	$8.8x10^{12}$
• DOSIMETRY POSITION IN REFLECTOR	260^{f}	$1410^{c}(\quad)^{d}$	126^{b}	1.9	$2.3x10^{12}$
• PVF (IN WATER)	260^{f}	$1621^{c}(\quad)^{d}$	94^{b}	2.9	$1.7x10^{11}$
• PVF (IN STEEL)	260	$1645^{c}(\quad)^{d}$	88^{b}	3.1	$1.5x10^{11}$
• PV - 1/4 T	260	$1723^{c}(\quad)^{d}$	75^{b}	3.6	$1.2x10^{11}$
• BETWEEN PV AND NEUTRON SHIELD TANK	60	$2314^{c}(\quad)^{d}$	50^{b}	6.1	$2.2x10^{10}$
FRANCE					
MELUSINE					
• PV-SIMULATOR (PCBT POSITION)	235 5	$1970(\quad)^{d}$		4 31	$1.8x10^{12}$
• CORE EDGE IN STEEL (61AVD)	235 5	$1450(\quad)^{d}$		2.13	$1.9x10^{13}$
• CORE EDGE IN STEEL (73 AVG)	235 5	$1450(\quad)^{d}$		2.14	$1.9x10^{13}$
GERMANY					
FRJ1					
• CORE EDGE	290 5	$\sim 1450(\quad)^{d}$		∼2	$2x10^{13}$
FRJ2					
• CORE (B5)	290 5	$1410(\quad)^{d}$	88	2.29	$4.8x10^{13}$
• REFLECTOR (2V4)	290 5	$1760(\quad)^{d}$	83	3.06	$3.7x10^{12}$
JAPAN					
JMTR					
• CORE POSITION IN STEEL	270-320				$1.0x10^{13}$
NETHERLANDS					
HFR (45 MW)					
• E5 CORE POSITION	250-650	1316^{g}	90	1.87	$2.1x10^{14}$

*4/12 Configuration (4cm H_2O between core and thermal shield and 12cm of H_2O between the thermal shield and PV wall).

(a) ASTM E693-79 iron (E), but using truncated spectra when the low energy (less than 0.1MeV spectrum) is not well defined by the calculations.

(b) ENDF/B-IV (E).

(c) Doran iron displacement cross section, HEDL SA-1058 (1976).

(d) With (n,) recoil displacements included. (UK used thermal flux mean energy at 260°C).

(e) Based on calculations BR3/core 4A.

(f) Temperature of water.

(g) Damage cross section library DANSIG 77, ECN-36 Report.

Cont'd

TABLE 3 (Cont'd)

TEST FACILITY	IRRADIATION TEMPERATURE (°C)	$\left[\frac{\overline{\sigma}\ \text{DPA}}{\phi(>1.0\text{MeV})}\right]$ (BARNS)	$\left[\frac{\overline{\sigma}\ \text{Fe}^{54}(n,p)}{\phi(>1.0\text{MeV})}\right]$ (MILLI-BARNS)	$\frac{(>0.1\text{MeV})}{(>1.0\text{MeV})}$	FLUX (>1.0MeV) LEVEL (n/cm -sec)
UNITED KINGDOM					
DIDO (25 MW)					
• CORE CENTRE (STEEL)	XXX±5	1620^a (1700)^d	110^b	2.3	5×10^{13}
• 2V4 POSITION (STEEL)	XXX±5	1700^a (2125)^d	130^b	3.8	3×10^{12}
PLUTO (25 MW)					
• CORE CENTRE (STEEL)	XXX±5	1620^a (1700)^d	110^b	2.3	5×10^{13}
• CA6B POSITION (STEEL)	XXX±5	1740^a (3480)^d	170^b	2.0	5×10^{11}
HERALD (5 MW)					
• CORE CENTRE (STEEL)	XXX±5	1410^a (1440)^d	100^b	1.9	3.5×10^{13}
• DIO POSITION (STEEL)	XXX±5	1510^a (1540)^d	80^b	2.8	3.5×10^{12}
UNITED STATES					
ORR-PSF (PV MOCKUP)* (30 MW)				(Peak)	
• SURVEILLANCE CAPSULE-CENTER	288±5	1510^a( )^d		2.2	8.8×10^{12}
• PV SURFACE POSITION-INCIDENT	"	1580^a( )^d		2.8	8.9×10^{11}
• PV 1/4T POSITION	"	1780^a( )^d		3.6	5.0×10^{11}
• PV 1/2T POSITION	"	2050^a( )^d		5.0	2.2×10^{11}
• PV VOID BOX BACK FACE POSITION	AMBIENT (44°C)	2520^a( )^d		6.9	2.9×10^{10}
ORR-BSR (30 MW - 2 MW)					(Maximum)
• CORE CENTER POSITION-IN FUEL	288±5	1410^a( )^d	~70^b	1.9	3.0×10^{13} 3.0×10^{14}
• SURFACE POSITION CENTER 4T HSST SPECIMEN	288±10	1640^a( )^d		3.1	4×10^{12}
BUFFALO (2 MW)					
• CORE POSITIONS IN NRL/NRC STEEL TEST ASSEMBLY	288±5	1750-1780^a ( )^d	69-77^b	3.41-3.67	3×10^{13}
VIRGINIA (2 MW)					
• CORE POSITION IN WEST/EPRI STEEL TEST ASSEMBLY	288±5	( )^d	~70^b	---	3×10^{13}
PWR					
• SURVEILLANCE POSITION	288-321			-	$0.1\text{-}2.0\times10^{11}$
• PV SURFACE	288-321				$0.07\text{-}0.70\times10^{11}$
• PV 1/4T POSITION	288-321	1660^a( )^d		3.1	$0.04\text{-}0.40\times10^{11}$
• EX-VESSEL CAVITY					$0.002\text{-}0.020\times10^{11}$
BWR					
• SURVEILLANCE POSITION (N/A)					
• PV SURFACE	280		200	~1.6	$10^8\text{-}10^9$
• PV 1/4T POSITION	170	1650^a( )^d		~2.50	$\sim10^8$
• EX-VESSEL CAVITY	56				$10^7\text{-}10^8$

TABLE 4

PROCEDURES FOR ANALYSIS AND INTERPRETATION OF NUCLEAR REACTOR SURVEILLANCE RESULTS

Step 1. Establish the basic surveillance test program for each operating power plant. Currently E185-79 is available and is used. However, updated versions of this standard should include the following:

a. Determination of surveillance capsule spatial flux-fluence-spectral and DPA maps for improved correlation and application of measured property change data (Upper shelf, ΔNDTT, etc.). Measured surveillance capsule fission and nonfission monitor reaction and reaction rate data should be combined with reactor physics computations to make necessary adjustments for capsule perturbation effects.

b. As appropriate, use of measured/calculated DPA damage for normalization of Charpy to Charpy (and other metallurgical specimen) variations in neutron flux, fluence, and spectra. Here, an increased use of a larger number of metallurgical specimen iron drillings may be appropriate for dosimetry.

Step 2. Establish a reactor physics computational method applicable to the surveillance program. Currently ASTM E482-76 and E560-77 provide general guidance in this area. However, updated versions of these standards should include the following:

a. Determination of corepower distributions applicable to long-term (30-40 year) irradiation. Associated with this is the need for the use of updated FSAR (Final Safety Analysis Report) reactor physics information at startup.

b. Determination of potential cycle-to-cycle variations in the core power distributions. This will establish bounds on expected differences between surveillance measurements and design calculations. Ex-vessel dosimetry measurements should be used for verification of this and the previous step.

c. Determination of the effect of surveillance capsule perturbations and photofission on the evaluation of capsule dosimetry. Adjustment codes should be used, as appropriate, to combine reactor physics computations with dosimetry measurements.

c. Benchmark validation of the analytical method.

Step 3. Establish methods for relating dosimetry, metallurgy, and temperature data from the surveillance program to current and future reactor vessel and support structure conditions. Currently E560-77 provides general guidance in this area. An updated version of this standard should include the following considerations.

a. Improved temperature monitoring.

b. Exposure units to be used to correlate observed changes in upper shelf and RT_{NDT} with neutron environment. This should lead to improved adjustments in trend curves for upper shelf and RT_{NDT}.

c. Differences in core power distributions which may be expected during long-term operation and which may impact the extrapolation of surveillance results into the future. As previously stated, ex-vessel dosimetry should be used for verification.

Step 4. Establish methods to verify steps 2 and 3 and to determine uncertainty and error bounds for the interpretation of the combined results of dosimetry, metallurgical and temperature measurements. Currently, ASTM E185-79 provides general guidance in this area. An updated version of this standard should more completely address the separate and combined accuracy requirements of dosimetry, metallurgy, and temperature measurement techniques.

TABLE 5

CHANGE IN DPA EXPOSURE FOR A FLUENCE OF 1.0 x 10^{19} n/cm^2 (E>1.0 MeV)

(In Going from PCA/PSF[a] Surveillance Capsule Location Through the Pressure Vessel Wall into the Ex-Vessel Void Box Cavity)

Position	Fluence (E > 1.0MeV) (n/cm^2)	DPA (DISP/Atom)[c]	$\frac{DPA(<1.0)}{DPA(Total)}$ [c]	$\frac{\phi(>0.1)}{\phi(>1.0)}$
Surveillance	1.0 x 10^{19}	.0151 (1.00)[b]	1.29	2.2
Incident PV	1.0 x 10^{19}	.0158 (1.05)	1.48	2.8
1/4T	1.0 x 10^{19}	.0178 (1.18)	1.73	3.6
1/2T	1.0 x 10^{19}	.0205 (1.36)	2.18	5.0
3/4T	1.0 x 10^{19}	.0243 (1.61)	2.71	6.5
In Cavity	1.0 x 10^{19}	.0252 (1.67)	2.88	6.9
PWR 1/4T	1.0 x 10^{19}	.0166 (1.10)	1.72	3.1
Test Reactor Core	1.0 x 10^{19}	.0141 (0.93)	1.20	1.9

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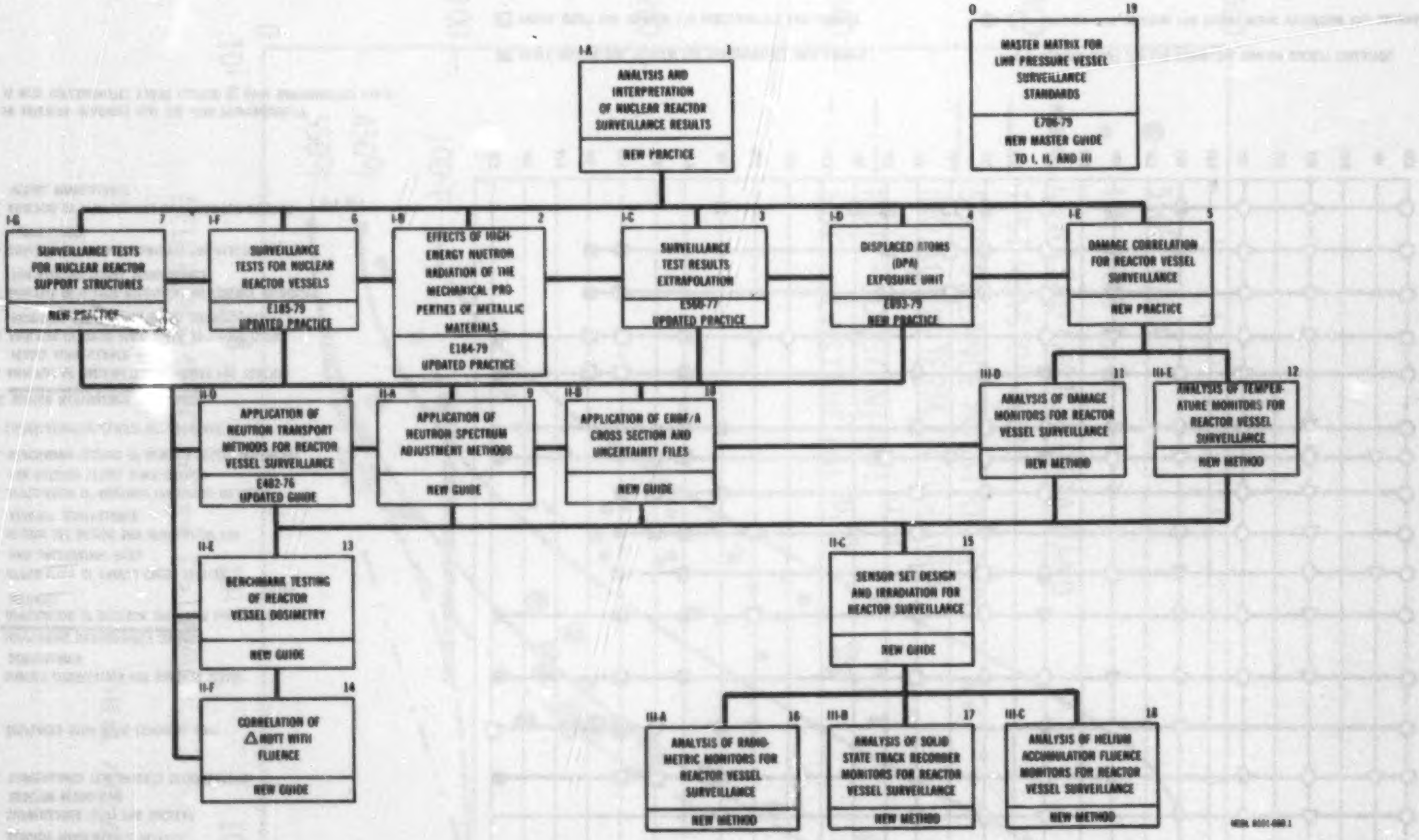
a) For the 12/13 configuration.
b) Normalization point.
c) For these calculations, the DPA cross section below 0.1 MeV has been set equal to zero because of uncertainties associated with the computation of the low energy flux. The actual DPA exposure, therefore, in the PV positions and in the cavity would be somewhat, but not significantly, higher.



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19
MASTER MATRIX FOR LWR PRESSURE VESSEL SURVEILLANCE STANDARDS
E706-79
NEW MASTER GUIDE TO I, II, AND III
I-A
1
ANALYSIS AND INTERPRETATION OF NUCLEAR REACTOR SURVEILLANCE RESULTS
NEW PRACTICE
I-G
7
SURVEILLANCE TESTS FOR NUCLEAR REACTOR SUPPORT STRUCTURES
NEW PRACTICE
I-F
6
SURVEILLANCE TESTS FOR NUCLEAR REACTOR VESSELS
E185-79
UPDATED PRACTICE
I-B
2
EFFECTS OF HIGH-ENERGY NEUTRON RADIATION OF THE MECHANICAL PROPERTIES OF METALLIC MATERIALS
E184-79
UPDATED PRACTICE
I-C
3
SURVEILLANCE TEST RESULTS EXTRAPOLATION
E560-77
UPDATED PRACTICE
I-D
4
DISPLACED ATOMS (DPA) EXPOSURE UNIT
E693-79
NEW PRACTICE
I-E
5
DAMAGE CORRELATION FOR REACTOR VESSEL SURVEILLANCE
NEW PRACTICE
II-D
8
APPLICATION OF NEUTRON TRANSPORT METHODS FOR REACTOR VESSEL SURVEILLANCE
E482-76
UPDATED GUIDE
II-A
9
APPLICATION OF NEUTRON SPECTRUM ADJUSTMENT METHODS
NEW GUIDE
II-B
10
APPLICATION OF ENDF/A CROSS SECTION AND UNCERTAINTY FILES
NEW GUIDE
III-D
11
ANALYSIS OF DAMAGE MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD
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ANALYSIS OF TEMPERATURE MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD
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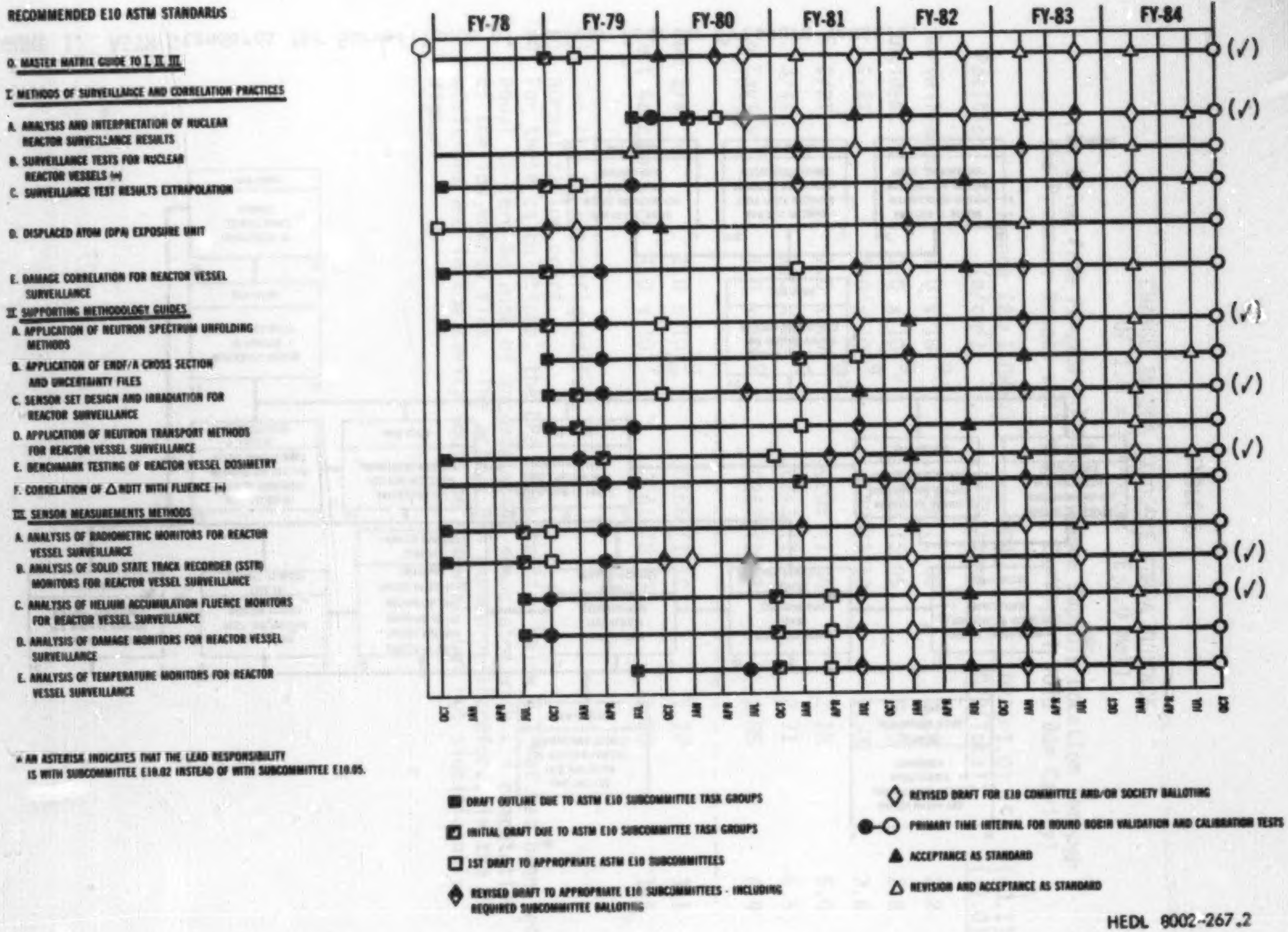
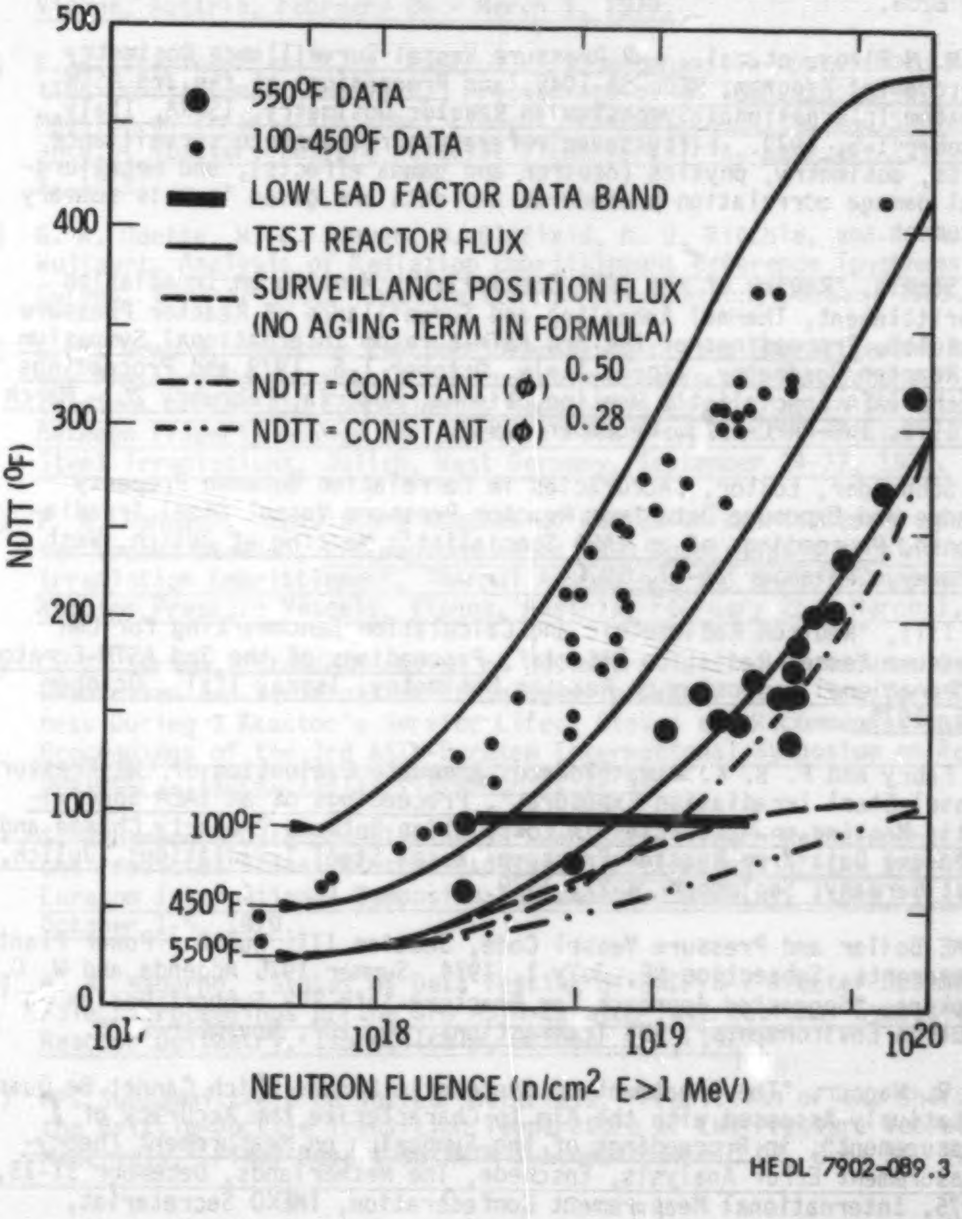
13
BENCHMARK TESTING OF REACTOR VESSEL DOSIMETRY
NEW GUIDE
II-F
14
CORRELATION OF △NDTT WITH FLUENCE
NEW GUIDE
II-C
15
SENSOR SET DESIGN AND IRRADIATION FOR REACTOR SURVEILLANCE
NEW GUIDE
III-A
16
ANALYSIS OF RADIOMETRIC MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD
III-B
17
ANALYSIS OF SOLID STATE TRACK RECORDER MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD
III-C
18
ANALYSIS OF HELIUM ACCUMULATION FLUENCE MONITORS FOR REACTOR VESSEL SURVEILLANCE
NEW METHOD

FIGURE 1. ASTM Standards for Surveillance of Nuclear Reactor Pressure Vessels.



HEDL 8002-267.2

FIGURE 2. Preparation, Validation, and Calibration Schedule for LWR Pressure Vessel Surveillance Standards. Neg 8010270-11



HEDL 7902-089.3

FIGURE 3. Comparison of Fluence Dependencies for the Shift in NDTT. Neg 7911456-4

REFERENCES

(1) ASTM Standard E185-79 gives reference to other standards and references that address the variables and uncertainties associated with property change measurements. The reference standards are A370, E8, E21, E23, and E208.

(2) W. N. McElroy, et. al., LWR Pressure Vessel Surveillance Dosimetry Improvement Program, HEDL-SA-1949, and Proceedings of the 3rd ASTM-Eruatom International Symposium on Reactor Dosimetry, ISPRA, Italy, October 1-5, 1979. Fifty-seven references relevant to surveillance tests, dosimetry, physics (neutron and gamma effects), and metallurgical damage correlation procedures and data are given in this summary document.

(3) L. Steele, "Review of the IAEA Specialist's Meeting on Irradiation Embrittlement, Thermal Annealing and Surveillance of Reactor Pressure Vessels", Proceedings of the 3rd ASTM-Euratom International Symposium on Reactor Dosimetry, Ispra, Italy, October 1-5, 1979 and Proceedings of the IAEA Specialist's Meeting, Vienna, Austria, February 26 - March 1, 1979, IWG-RRPC-79/3, December, 1979.

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APPENDIX B

STANDARD RECOMMENDED PRACTICE FOR ANALYSIS AND INTERPRETATION OF LIGHT WATER REACTOR SURVEILLANCE RESULTS [E706(IIIA)]

REFERENCES (Cont'd)

(18) F. Adams and R. Dams, Applied Gamma-Ray Spectrometry, Pergamon Press, New York, N.Y., 1970.

(19) R. Gold, R. J. Armani, and J. H. Roberts, "Absolute Fission Rate Measurements with Solid State Track Recorders," NSE 34, 13-32, 1968.

(20) W. [illegible] and [illegible], "Fission Product [illegible] Measurements for Neutron Dosimetry and Damage Correlations", NUREG/CP-0004, Vol. 2, 725, Oct. 1977.

STANDARD RECOMMENDED PRACTICE FOR ANALYSIS AND INTERPRETATION OF LIGHT WATER REACTOR SURVEILLANCE RESULTS [E706(IIIA)]

1.0 SCOPE

1.1 This recommended practice describes the methodology to be used in the analysis and interpretation of data obtained from LWR pressure vessel surveillance programs; and, based on the results of that analysis, establishes a formalism to be used to evaluate present and future condition of the pressure vessel. The procedures, broken down into four steps, are summarized in Table 1.

1.2 This practice relies on, and ties together, the application of several supporting ASTM recommended practices, methods, and guides which are in various stages of completion (see Figure 1 and ASTM Standard E706). In order to make this practice at least partially self-contained, a moderate amount of discussion is provided in areas relating to the ASTM documents.

1.3 Support subject areas which are discussed include reactor physics calculations, dosimeter selection and analysis, exposure units, and temperature correlations.

1.4 This practice is restricted to direct applications related to surveillance programs which are established in support of the operation, licensing, and regulation of LWR nuclear power plants. Procedures and data related to the analysis, interpretation, and application of test reactor results are addressed in ASTM Standards E706 (IE), Damage Correlation for Reactor Vessel Surveillance and E706 (IC), Surveillance Test Results Extrapolation.

1.5 The related area of radiation damage to reactor vessel support structures involves the application of detailed neutron transport calculations of the reactor cavity as well as the use of low temperature damage correlations and is, therefore, not specifically addressed in this practice. Procedures applicable to reactor vessel support structure analysis are described in ASTM Standard E706 (IG), Surveillance Tests for Nuclear Reactor Vessel Support Structures.

2.0 APPLICABLE DOCUMENTS

2.1 ASTM Standards

E706 (O)	Master Matrix for LWR-PV Surveillance Standards
E706 (IB), E184	Effects of High Energy Neutron Radiation on the Mechanical Properties of Metallic Materials
E706 (IC), E560	Surveillance Test Results Extrapolation
E706 (ID), E693	Displaced Atoms (DPA) Exposure Unit
E706 (IE)	Damage Correlation for Reactor Vessel Surveillance
E706 (IF), E185	Surveillance Tests for Nuclear Reactor Vessels
E706 (IG)	Surveillance Tests for Nuclear Reactor Vessel Support Structures
E706 (IIA)	Application of Neutron Spectrum Adjustment Methods
E706 (IIB)	Application of ENDF/A Cross Section and Uncertainty Files
E706 (IIC)	Sensor Set Design and Irradiation for Reactor Surveillance
E706 (IID), E482	Application of Neutron Transport Methods for Reactor Vessel Surveillance
E706 (IIE)	Benchmark Testing of Reactor Vessel Dosimetry
E706 (IIF)	Correlation of ΔNDTT with Fluence
E706 (IIIA)	Analysis of Radiometric Monitors for Reactor Vessel Surveillance
E706 (IIIB)	Analysis of Solid State Track Recorder Monitors for Reactor Vessel Surveillance
E706 (IIIC)	Analysis of Helium Accumulation Fluence Monitors for Reactor Vessel Surveillance
E706 (IIID)	Analysis of Damage Monitors for Reactor Vessel Surveillance
E706 (IIIE)	Analysis of Temperature Monitors for Reactor Vessel Surveillance

2.2 NUREG/CR-0038, -0127, -0285, -0550, -0551, -0720, -1240 Vol 1, 2, 3, -1241, Vol 1--NRC LWR Surveillance Dosimetry Quarterly Reports

NRC NUREGXXXX Special Publication, PCA Experiment and Blind Test

2.3 ASME Standards

ASME III Boiler and Pressure Vessel Code

Code of Federal Regulations 10 CFR Part 50, Appendix G

Code of Federal Regulations 10 CFR Part 50, Appendix H

2.4 References 1 to 20

3.0 SIGNIFICANCE

3.1 The Code of Federal Regulations (10CFR Part 50, Appendix G) requires the implementation of a reactor vessel materials surveillance program for all operating LWRs. The stated objectives of the surveillance program are twofold. The first requirement of the program is "to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment". The second requirement is to make use of the data obtained from the surveillance program "to determine the conditions under which the vessel can be operated with adequate margins of safety throughout its service life".

3.2 To satisfy the first requirement of Section 3.1, the tasks to be carried out are straightforward. Each of the irradiation capsules which comprise the surveillance program may be treated as a separate experiment. The goal then is to define and carry to completion a dosimetry program which will a posteriori describe the neutron field to which the materials test specimens were exposed. The resultant information will then become a part of a data base applicable in a stricter sense to the specific plant from which the capsule was removed, but also in a broader sense to the industry as a whole.

3.3 To satisfy the second requirement of Section 3.1 the tasks to be carried out are somewhat complex. Here the objective is to accurately describe the neutron field to which the pressure vessel itself will be exposed over its service life. This description of the neutron field must include spatial flux gradients within the vessel wall. Therefore, heavy emphasis must be placed on the use of neutron transport techniques as well as on the choice of a design basis for the computations. Since a given surveillance capsule measurement, particularly those obtained early in plant life, is not necessarily representative of long term reactor operation, a simple normalization of neutron transport calculations to dosimetry data from a given capsule may not be appropriate.

4.0 ESTABLISHMENT OF THE SURVEILLANCE PROGRAM

4.1 ASTM E185 describes the criteria that should be considered in planning and implementing surveillance test programs and points out precautions that should be taken to ensure that: (1) capsule exposures can be related to beltline exposures, (2) materials selected for the surveillance program are samples of those materials most likely to limit the operation of the reactor vessel, and (3) the tests yield results useful for the evaluation of radiation effects on the reactor vessel

4.2 From the viewpoint of the radiation analyst, the criteria set forth in ASTM E185 are met by the completion of the following tasks:

(1) Determine the locations within the reactor which provide suitable lead factors (see 10CFR50, Appendix G) for each irradiation capsule relative to the pressure vessel.

(2) Select neutron sensor sets which provide adequate coverage over the neutron and fluence ranges of interest.

(3) Specify sensor set locations within each irradiation capsule to define neutron field gradients within the metallurgical specimen array.

(4) Establish an adequately benchmarked neutron transport methodology to be used both in the analysis of individual sensor sets and in the projection of materials properties changes to the vessel itself.

4.3 The first three items listed in the preceding paragraph usually fall under the jurisdiction of the reactor vendor. However, the fourth item, which directly addresses the analysis and interpretation of surveillance results, could be performed by any number of independent service laboratories. It is recommended, therefore, that the documentation describing the surveillance programs of individual reactors provide details of irradiation capsule construction and sensor set design adequate to perform independent evaluations of surveillance measurements.

4.4 Information regarding the choice of neutron sensor sets for LWR surveillance applications is provided in ASTM E706 (IIC), "Sensor Set Design" and E706 (III A), "Radiometric Monitors". Foil materials currently in common usage and acceptable for use in surveillance programs include Cu, Ti, Fe, Ni, U^{238}, Np^{237}, U^{235}, and Co-Al. All sensor sets should be calibrated to known counting standards such as those supplied by NBS or IAEA. All quality assurance information pertinent to the sensor sets must be documented with the description of the surveillance program.

4.5 As indicated in paragraph 4.2, neutron transport methods are used both in the design of the surveillance program and in the analysis and interpretation of capsule measurements. During the design phase, neutron transport calculations are used to define the neutron field within the pressure vessel wall and, in conjunction with damage trend curves, to predict the degree of embrittlement of the reactor vessel over its service life. Embrittlement gradients are in turn used to determine pressure temperature limitations for normal plant operation as well as to evaluate the effect of various heatup/cooldown transients on vessel condition.

4.6 The neutron transport methodology used for these computations must be well benchmarked and qualified for application to LWR configurations. The PCA/PSF* data documented in the NUREGs, Section 2.2, provide an excellent configuration for benchmarking basic transport methodology and input data. However, further analytical/experimental comparisons are required to qualify a method for application to LWRs which have a more complex geometry than the PCA/PSF mockups. This additional qualification may be achieved by comparison with surveillance capsule measurements and/or comparison with measurements taken in the reactor cavity external to the pressure vessel of selected operating reactors.

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* LWR Pressure Vessel mockups established in the ORNL Pool Critical Assembly (PCA) and in the Pool Side Facility (PSF) of the Oak Ridge Research Reactor.

Benchmarking and qualification will be considered sufficient when calculated values for the Fe^{54} (n,p) Mn^{54}, Ni^{58} (n,p) Co^{58}, ${}_{U}^{238}$(n,f), and N_p^{237} (n,f) reaction rates agree with measurements within ± 15%.(7). All experimental/analytical comparisons which comprise the qualification program for a neutron transport methodology must be documented.

4.7 Since neutron transport calculations for LWRs are intended to be applicable for long term irradiations, the design basis core power distributions should be representative of time averaged conditions for the generic reactor type in question. Computations performed with this design basis will provide a baseline against which all future surveillance capsule measurements will be compared; and will provide the vehicle by which pressure vessel embrittlement will be related to a body of surveillance capsule data. Further discussions of the application of neutron transport methods to LWRs are provided in E706 (IC) "Surveillance Dosimetry Extrapolation" and E706 (IID) "Application of Neutron Transport Methods."

4.8 To assure that metallurgical results obtained from surveillance capsule measurements may be applied to the determination of the pressure vessel fracture toughness, the irradiation temperature of the surveillance test specimens must be monitored. Methodologies to be used for temperature monitoring during long term irradiations in an LWR environment are discussed in E706 (III E) "Temperature Monitors."

5.0 ANALYSIS OF INDIVIDUAL SURVEILLANCE CAPSULES

5.1 It is recognized that for many operating power reactors the documentation of baseline neutron transport calculations and sensor set design information may not be available. In that event, the required information should be provided by the service laboratory in the respective surveillance report.

5.2 Radiometric analysis of capsule sensor sets should follow procedures outlined in ASTM E706. For sensors such as the fission monitors which may be gamma ray sensitive, photo reaction corrections should be derived from the results of gamma ray transport calculations performed for the explicit capsule configuration under examination. Photo reaction corrections in LWR environments have been shown to be extremely configuration dependent.

5.3 In calculating spectrum averaged reaction cross-sections from neutron transport calculations, care should be taken to model the explicit capsule configuration under examination (see E706 (IID)). Spectrum adjustment and uncertainty analysis for the determination of damage exposure parameters should be done according to the procedures outlined in E706 (IIA) "Application of Neutron Spectrum Adjustment Methods". To achieve satisfactory uncertainty bounds for the damage parameters a sufficiently large set of foils should be used as stipulated in Sections 4.2 and 4.4.

5.4 The report of the capsule analysis should contain the information listed below. The estimated accuracy (uncertainties and errors) for all data should be defined.

(1) Damage exposure parameters at the position of the metallurgical specimens. These values will be used for correlation with metallurgical data to develop damage trend curves. Neutron fluence (E>1.0 Mev) is presently required. However, dpa (displacements per atom) and neutron fluence (E>0.1 Mev) should also be included for future reference. These exposure values are derived from a combination of measurements and calculations and must include estimates of uncertainty and error bounds.

(2) The neutron spectra, reaction rates, reaction cross-sections, and all other nuclear constants used in the derivation of exposure values for the capsule.

(3) The gamma ray energy spectra and reaction cross-sections used to make photoreaction corrections for the neutron sensor sets.

(4) The power-time history of the reactor during the irradiation period of the subject capsule.

(5) Spatial gradients of neutron flux ($E>1.0$ Mev), neutron fluence ($E>1.0$ Mev), and dpa throughout the metalurgical specimen array.

In addition, the documentation supporting the benchmarking/qualification of sensor sets and reactor physics methodology should be either referenced or included as an appendix to the dosimetry report.

6.0 PROTECTION OF VESSEL CONDITION FOR FUTURE PLANT OPERATION

6.1 This recommended practice requires the use of a fully benchmarked and qualified neutron transport methodology in both the design of the surveillance program and the analysis of individual surveillance capsules. The neutron field information obtained from these computations should also be used to project damage gradients within the pressure vessel wall. Currently 10 CFR Part 50, Appendices G and H and Regulatory Guide 1.99 require that all such projections be based on neutron fluence ($E>1.0$ Mev). However, it is recommended that supplementary projections based on dpa maps throughout the pressure vessel/surveillance capsule geometry be included in the surveillance report.

6.2 If the qualified reactor physics computations have used time averaged core power distributions as a design basis, the analytical results will have an advantage over any single surveillance capsule measurement which, at least early in plant life, has been influenced by core power distributions that may differ by up to ± 15% from long term average distributions. Therefore, normalization of neutron transport calculations to a single surveillance capsule measurement may well result in larger inaccuracies than are introduced by using the analytical results directly.

6.3 It is recommended that all surveillance results for a generic reactor type (similar reactor geometry and fuel loading) be used as a data base to qualify the reactor physics methodology as to its applicability to a particular reactor system. This approach should, in the long term, provide a statistically significant validation of the calculations. Moreover, evaluation of a relatively large number of surveillance capsules removed after extended irradiation periods should also verify the time averaged power distributions used in the analysis. In essence, capsules removed from geometrically symmetric positions in generic reactors represent a series of repeat measurements. As such, the measured data will reflect the variability in important parameters such as water temperature, reactor dimensions, fuel loading, sensor set design, sensor set analysis, and reactor operating characteristics. By taking advantage of a large data base obtained from these repeat measurements, the uncertainties introduced by these variable parameters may be better understood and possibly reduced.

6.4 When evaluating the results of a given surveillance capsule analysis, the measured capsule exposure should be compared directly with neutron transport analysis and with all available experimental data obtained from similar capsules removed from reactors having the same design. If the agreement between measurement and calculation is within a range of ± 15%, the analytically derived neutron field parameters should be used for all damage determinations for the pressure vessel.

6.5 If the measurements differ from the calculations by more than 15%, further investigation of the measurement approach and the mode of operation of the reactor in question should be undertaken. Any adjustments made to vessel embrittlement projections based on the results of these investigations should be justified and fully documented in the surveillance report.

7.0 ACCURACIES

7.1 Analysis and measurement accuracies (uncertainties and errors) in the areas of concern for this practice may be difficult to determine. However, they should be properly addressed. When uncertainties and errors are well defined, as in integral reaction rate measurements, they should be estimated and summarized in an accuracy table. For more difficult kinds of uncertainties and errors (as in spectrum adjustment), a state-of-art analysis will have to be chosen and a statement must be included which indicates what the uncertainty and error estimates do and do not cover. It will be necessary to accept incomplete or non-rigorous uncertainty and error estimated when there is no readily available alternative.

TABLE 1

PROCEDURES FOR ANALYSIS AND INTERPRETATION OF NUCLEAR REACTOR SURVEILLANCE RESULTS

Step 1. Establish the basic surveillance test program for each operating power plant. Currently E185-79 is available and is used. However, updated versions of this standard should include the following:

a. Determination of surveillance capsule spatial flux-fluence-spectral and DPA maps for improved correlation and application of measured property change data (Upper shelf, ΔNDTT, etc.). Measured surveillance capsule fission and nonfission monitor reaction and reaction rate data should be combined with reactor physics computations to make necessary adjustments for capsule perturbation effects.

b. As appropriate, use of measured/calculated DPA damage for normalization of Charpy to Charpy (and other metallurgical specimen) variations in neutron flux, fluence, and spectra. Here, an increased use of a larger number of metallurgical specimen iron drillings may be appropriate for dosimetry.

Step 2. Establish a reactor physics computational method applicable to the surveillance program. Currently ASTM E482-76 and E560-77 provide general guidance in this area. However, updated versions of these standards should include the following:

a. Determination of core power distributions applicable to long-term (30-40 year) irradiation. Associated with this is the need for the use of updated FSAR (Final Safety Analysis Report) reactor physics information at startup.

b. Determination of potential cycle-to-cycle variations in the core power distributions. This will establish bounds on expected differences between surveillance measurements and design calculations. Ex-vessel dosimetry measurements should be used for verification of this and the previous step.

c. Determination of the effect of surveillance capsule perturbations and photofission on the evaluation of capsule dosimetry. Adjustment codes should be used, as appropriate, to combine reactor physics computations with dosimetry measurements.

d. Benchmark validation of the analytical method.

Step 3. Establish methods for relating dosimetry, metallurgy, and temperature data from the surveillance program to current and future reactor vessel and support structure conditions. Currently E560-77 provides general guidance in this area. An updated version of this standard should include the following considerations.

a. Improved temperature monitoring.

b. Exposure units to be used to correlate observed changes in upper shelf and RT_{NDT} with neutron environment. This should lead to improved adjustments in trend curves for upper shelf and RT_{NDT}.

c. Differences in core power distributions which may be expected during long-term operation and which may impact the extrapolation of surveillance results into the future. As previously stated, ex-vessel dosimetry should be used for verification.

Step 4. Establish methods to verify steps 2 and 3 and to determine uncertainty and error bounds for the interpretation of the combined results of dosimetry, metallurgical and temperature measurements. Currently, ASTM E185-79 provides general guidance in this area. An updated version of this standard should more completely address the separate and combined accuracy requirements of dosimetry, metallurgy, and temperature measurement techniques.

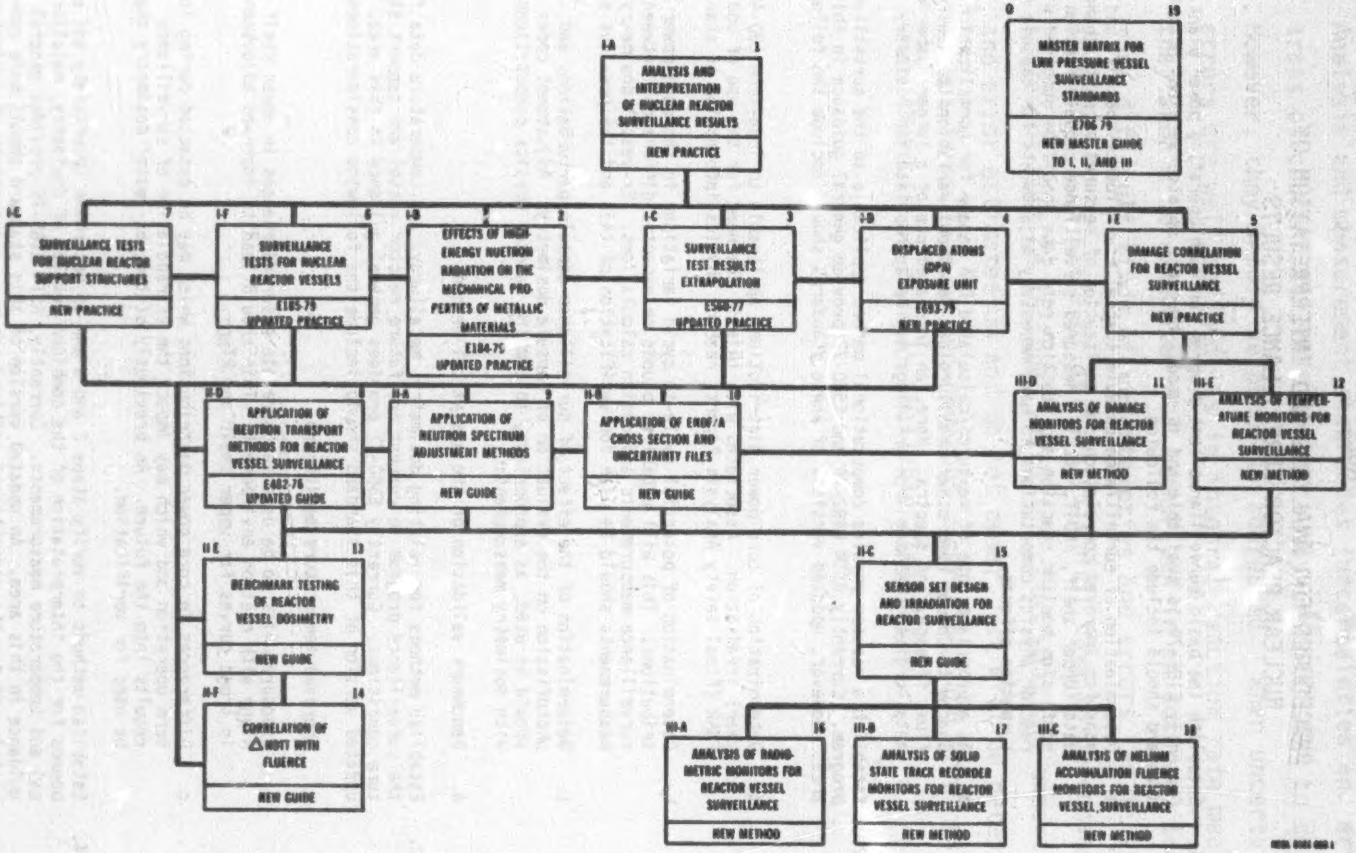
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FIGURE 1. ASTM Standards for Surveillance of Nuclear Reactor Pressure Vessels.

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This report describes progress made in the Light water Reactor Pressure Vessel Surveillance Dosimetry Improvement program during 1980. The primary concern of this program is to improve, standardize, and maintain dosimetry, damage correlation, and the associated reactor analysis procedures used for predicting the integrated effect of neutron exposure to LWR pressure vessels and support structures.

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